Advanced Reactors Spent Fuel and Waste Streams Disposition Strategies

Spent Fuel and Waste Disposition

Prepared for
US Department of Energy
Spent Fuel and Waste Science and Technology

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APPENDIX E
NFCSC DOCUMENT COVER SHEET [1]

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This deliverable was prepared in accordance with Sandia National Laboratories
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QA program which meets the requirements of

☒ DOE Order 414.1 ☐ NQA-1 ☐ Other

This Deliverable was subjected to:

☒ Technical Review ☐ Peer Review

Technical Review (TR)
Review Documentation Provided

☐ Signed TR Report or,
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☒ Signature of TR Reviewer(s) below

Peer Review (PR)
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Name and Signature of Reviewers

Phillip Honnold

NOTE 1: *Appendix E should be filled out and submitted with the deliverable. Or, if the PICS:NE system permits, completely enter all applicable information in the PICS:NE Deliverable Form. The requirement is to ensure that all applicable information is entered either in the PICS:NE system or by using the NTRD Document Cover Sheet.*

- *In some cases there may be a milestone where an item is being fabricated, maintenance is being performed on a facility, or a document is being issued through a formal document control process where it specifically calls out a formal review of the document. In these cases, documentation (e.g., inspection report, maintenance request, work planning package documentation or the documented review of the issued document through the document control process) of the completion of the activity, along with the Document Cover Sheet, is sufficient to demonstrate achieving the milestone.*

EXECUTIVE SUMMARY

This report describes research and development (R&D) activities conducted during Fiscal Year 2023 (FY23) in the Advanced Fuels and Advanced Reactor Waste Streams Strategies work package in the Spent Fuel Waste Science and Technology (SFWST) Campaign supported by the United States (U.S.) Department of Energy (DOE). This report is focused on evaluating and cataloguing Advanced Reactor Spent Nuclear Fuel (AR SNF) and Advanced Reactor Waste Streams (ARWS) and creating Back-end Nuclear Fuel Cycle (BENFC) strategies for their disposition. The R&D team for this report is comprised of researchers from Sandia National Laboratories and Enviro Nuclear Services, LLC.

High-level Purpose of this work: The commercial nuclear reactor industry is developing advanced reactor (AR) fuel cycles and advanced fuel (e.g., accident tolerant fuel; ATF) for the existing fuel cycle. Waste streams from ARs span a variety of fuel types and waste types, some of which are similar to existing spent nuclear fuels (SNF) from previous reactors (e.g., TRISO particle fuels and metallic fuels within the DOE-managed SNF (DSNF) inventory) and some of which may be different (e.g., molten salt reactor fuels). The purpose of this work is to identify gaps and outline areas where further research would contribute to a well-defined disposition pathway for SNF from AR designs and their associated potential additional waste streams. The analysis of disposition pathways is rooted in characterization of potential waste forms and the precedent BENFC experience that can inform approaches to disposition of ARWS and ARF (see Swift and Sassani, 2020, and Sassani et al., 2022 for examples).

This report develops a high-level strategy via accomplishment of three primary objectives:

(1). *Survey the range of AR SNF and associated waste streams in order to classify the AR fuels and wastes into groups to be evaluated.* To the extent possible, this includes understanding implementation of AR concepts such that additional waste streams associated with the AR fuel cycle can also be taken into consideration.

(2). *Collate and evaluate the previous experience on existing analogous SNF* and wastes to build a knowledge base for evaluating storage, transportation, and disposal issues/gaps related to the AR SNF and waste streams.

(3). *Identify technical gaps related to storage, transportation, and disposal of AR fuels and ARWS.* Evaluate AR fuels and Advanced Reactor Waste Streams (ARWS) against precedent experience in the knowledge base to produce a detailed gaps analysis (for S&T R&D), and analyses of features, events, and processes (FEP, for disposal R&D). Such gap and FEP analyses can be used to identify additional R&D activities in order to close the gaps.

Scope: The focus of this report is on the disposition of AR spent fuels and ARWS. ATF is discussed for two primary reasons: 1) the study of differences in disposition between of ATF and traditional light water reactor (LWR) fuel offers insights to areas of concern that overlap with AR fuel, and 2) some advanced LWR are typically classified as AR's and may utilize ATF. Small modular reactors (SMRs) and microreactors are an emerging subset of AR concepts that are being developed, but in general some of these have spent fuel very similar to most current LWR fuels with some minor differences (e.g., shorter in length) which generally do not appear to introduce conceptual differences in storage, transportation, and disposal of those AR SNF. For other SMR with fuels similar to those considered for larger AR (e.g., molten salt reactors), those aspects of the BENFC are covered by those aspects. However, there is no current precedent for storage/transport/disposal of a small-modular-/micro-reactor in total, and such a gap currently is outside the scope included in this report.

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Accomplishments: This report summarizes the analyses conducted by the coauthors.

- A subset of commercial advanced reactor concepts have been reviewed and summarized.
- The AR SNF and potential other waste streams (ARWS) from these reactor designs have been reviewed and summarized.
- Previous BENFC experience and precedent pathways for disposition (storage, transportation, and disposal) of similar SNF and wastes have been reviewed and summarized.
- Preliminary technical gaps between previous BENFC experience/precedents and ARF and ARWS have been identified and summarized.
- The strategy and approaches to the detailed gap analyses and FEP analyses have been developed.

Key Findings:

- A review of Regulatory Considerations was conducted. Given the differences of AR SNF and AR waste streams from the characteristics of Light Water Reactor (LWR) SNF, updates or supplements to methodology/guidance may facilitate application of existing regulations to evaluations for AR SNF and ARWS.
- The existing *regulatory guidance* is largely based on the low-enriched zircaloy clad uranium-oxide SNF. Updates to guidance may facilitate the application of existing regulations to address BENFC assessments for ARF and ARWS.
- Uncertainty around the specifics of AR designs can be the result of specifications being considered proprietary information or because the concept is still at a low level of technical readiness. This uncertainty presents difficulties to conduct full and thorough of analysis of ARF and ARWS disposition.
- Based on a survey of reactor types, three fuel types are most common: TRISO, Sodium-bonded Metallic, and Salt Fuel. Of these three, literature review and analysis indicate that TRISO is the only AR Fuel with a clear pathway for direct disposal, while salt fuel waste is possible in certain repository design concepts. Sodium-bonded metallic fuels must be processed and/or treated, as the metallic sodium bond reacts violently with water, which presents hazards not only for disposal, but can also present potential issues for storage and transportation.
- While TRISO fuel has a clear direct disposition pathway, it should be noted that previous BENFC experience with TRISO fuel is related to prismatic TRISO from the Ft. St. Vrain reactor, and some emerging commercial concepts employ TRSIO pebbles in a Pebble Bed Reactor (PBR). Fuel compacts removed from the prismatic block may necessitate further analysis, especially with respect to safeguards and security and packaging and handling. TRISO from a molten salt-cooled PBR may also experience an ingress of salt into the graphite/coated particle matrix, so additional investigation may be indicated in this area.
- Many commercial designs plan to use High Assay Low-Enriched Uranium (HALEU) fuel (enriched between 5 and 19.75%), which will generally be used to a higher burn-up than traditional light-water reactor (LWR) fuels. This has implications across the BENFC in storage, transportation, and disposal, as higher thermal loads tend to accelerate degradation mechanisms. There are also notable implications for criticality safety associated with HALEU.
- Accident tolerant fuels (ATFs) are generally expected to be as, or somewhat more, robust, as standard LWR fuels in the disposal environment. However, thermal and mechanical properties of ATFs are sufficiently different such that further analyses are warranted, especially for impacts on

storage and transportation. The additional elements added to ATF have potential to affect near-field geochemistry in the disposal setting. Because there is a variety of ATF, scoping studies will be done on a case-by-case basis to assess the BENFC.

ACKNOWLEDGEMENTS

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ACRONYMS / ABBREVIATIONS

AEA	Atomic Energy Act
AF	advanced fuel(s)
AR	advanced reactor(s)
AR Fuels	advanced reactor fuels
ARWS	advanced reactor waste stream(s)
ATF	accident tolerant fuel(s)
BENFC	back end of the nuclear fuel cycle
BWR	boiling water reactor
CWF	ceramic waste form
DCSS	dry cask storage system(s)
DOE	U.S. Department of Energy
DPC	dual-purpose canister
EBS	engineered barrier system
EMT	Electrometallurgical Treatment
FEP	Features, Events, and Processes
FIMA	Fissions per Initial Metal Atom
GTCC	Greater-Than-Class C
HAC	hypothetical accident condition(s)
HALEU	High-Assay Low-Enriched Uranium
HEU	highly-enriched uranium
HLW	high-level waste
HTGR	high temperature gas-cooled reactor
INL	Idaho National Laboratory
ISFSI	Independent Spent Fuel Storage Installation
LANL	Los Alamos National Laboratory
LEU	low-enriched uranium
LLW	low-level waste
LOCA	loss-of-coolant accident
LWR	light water reactor
MC&A	material control and accounting
MSR	Molten salt reactor
MSRE	Molten-Salt Reactor Experiment
MTHM	metric tons of heavy metal

NE	Nuclear Energy
NRC	U.S. Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission designation for regulations
NWPA	Nuclear Waste Policy Act
ORNL	Oak Ridge National Laboratory
PBMR	pebble bed modular reactor
PBR	pebble bed reactor
PMR	prismatic block reactor
PWR	pressurized water reactor
RCRA	Resource Conservation and Recovery Act
R&D	research and development
RGs	regulatory guidance
SAR	Safety analysis report
SFP	spent fuel pool(s)
SFR	sodium-cooled fast reactor
SFWST	Spent Fuel and Waste Science and Technology
SNF	spent nuclear fuel
SRP	Standard Review Plan
S&T	storage and transport
TRISO	TRi-structural ISOtropic
TRU	transuranic elements
UN	uranium nitride
VHTR	very high temperature gas–cooled reactor
WF	waste form(s)
WG	waste group(s)
YMP	Yucca Mountain Project

ADVANCED REACTORS SPENT FUEL AND WASTE STREAMS DISPOSITION STRATEGIES

This report fulfills milestone report M2SF-23SN010102015 in the Spent Fuel and Waste Science and Technology (SFWST) work package (SF-23SN01010201). This work was sponsored under the Department of Energy's (DOE) Office of Nuclear Energy (NE) Spent Fuel and Waste Disposition campaign.

1 INTRODUCTION

The commercial nuclear reactor industry is developing advanced reactors (AR) for new fuel cycles and advanced fuels (e.g., accident tolerant fuels; ATF) for the existing fuel cycle. Waste streams from AR span a variety of fuel types and waste types, some of which are similar to existing spent nuclear fuels (SNF) from previous reactors (e.g., TRISO particle fuels and metallic fuels from DOE-managed SNF (DSNF) and some of which are potentially different (e.g., molten salt reactor fuels).

The report presents a high-level strategy for the back end of the nuclear fuel cycle (BENFC) considerations for 1) potential fuels and waste streams from future advanced reactor fuel cycles and 2) the newer fuels in the current light water reactor (LWR) fuel cycle (e.g., ATF). The strategy outlines a path to the dispositioning of AR waste streams for their storage, transportation, and disposal considerations, with a goal of leveraging previous work including that from the YM SAR (DOE, 2008) and the disposal options evaluation (SNL 2014; see Swift and Sassani, 2020, and Sassani et al., 2022 for examples), as well as storage and transportation R&D on current LWR fleet spent fuels. Given the additional work identified within the most recent S&T Gap Analysis (Honnold *et al*., 2021) that covers ATF and some potential AR fuels (e.g., TRISO), the approach in this study aims to develop the higher-level summary and strategy across the entire back end of the nuclear fuel cycle (BENFC) for advanced fuels.

This work develops a high-level strategy via three primary objectives:

(1). *Survey the range of AR SNF and associated waste streams in order to classify the AR fuels and wastes into groups to be evaluated.* To the extent possible, this includes understanding implementation of AR concepts such that additional waste streams associated with the AR fuel cycle can also be taken into consideration.

(2). *Collate and evaluate the previous experience on existing analogous SNF* and wastes to build a knowledge base for evaluating storage, transportation, and disposal issues/technical gaps related to the AR SNF and waste streams.

(3). *Identify technical gaps related to storage, transportation, and disposal of AR SNF and ARWS.* Evaluate AR fuels and Advanced Reactor Waste Streams (ARWS) against precedent experience in the knowledge base to produce a detailed gaps analysis. This technical gap analysis can be used to identify additional R&D activities in order to close the gaps.

Technical readiness of AR concepts is an important consideration in the development of the SNF and waste dispositioning strategy. It seems sensible to assign higher priority for analysis to the more mature concepts that have higher probability of deployment in the near term. Additionally, because a mature concept has likely committed to specifics of the AR concept implementation, specific details (e.g., fuel type and fuel operating parameters), are more readily available than would be the case for less mature AR concepts. Advanced reactors that are at more advanced stages of design/development will have the needed information/data for detailed analyses and gap identification relative to existing commercial SNF and DSNF. Less advanced AR concepts are still covered at the conceptual level to identify major

potential gaps given their AR SNF and other waste streams. In this report, the waste streams with existing BENFC information are identified, and detailed summaries created for them. For less-known waste streams, additional considerations are identified to begin the process of building an R&D foundation for disposition strategies.

1.1 Report Overview and Methodology

1.1.1 Overview

This report is arranged as follows:

- Chapter 2 provides a review of the Regulatory Considerations that may come into play in the development of disposal pathways for ARF and ARWS. Topics covered include Storage, Transportation, Disposal, and Siting/Construction/Operation/Decommissioning. The existing regulatory framework for all of the above areas is discussed, as well as ways in which the regulatory framework may or may not cover certain aspects of AR SNF and ARWS.
- Chapter 3 provides an overview of commercial concepts and survey of the major potential AR SNF and ARWS, including preliminary gap assessment for select AR concepts.
- Chapter 4 provides an overview of Back-end Nuclear Fuel Cycle (BENFC) Challenges.
- Chapter 5 presents a review of the relevant precedent pathways for Storage, Transport, and Disposal. Particular attention is given to previous analyses of direct disposal for AR SNF, as well as potential treatment options for those fuels that are potentially unsuitable for direct disposal.
- Chapter 6 presents conclusions, summarizes preliminary technical gaps, and identifies potential R&D to close these gaps.

1.1.2 Methodology

The methodology of this report is to first review the characteristics of AR SNF and Advanced Reactor Waste streams (ARWS), with a focus on Storage, Transportation, and Disposal. Regulatory Considerations are discussed in the opening chapter to provide the context for performing gap/FEP analyses for a particular AR SNF waste category for storage, transportation, and disposal comprising the BENFC. In some cases, AR SNF and ARWS can be categorized within an existing category (or, in some cases, categories) (see SNL 2014 for in-depth discussion for Waste Categorization). Identifying waste characteristics determines the initial bases for evaluating waste disposition based upon existing analogous DSNF, DOE-managed High-Level Waste (DHLW) and previous experience with their disposition. In this way, gaps and challenges can be identified via a comparison of AR SNF or ARWS to existing analogous waste with which it shares some subset of characteristics. Also, identifying how the potential AR SNF and ARWS are different than the existing analogs allows identification of potential gaps for further R&D.

Categorizing nuclear fuel cycle waste by waste type or waste group is a practical methodology that allows for grouping wastes according to their characteristics, especially those characteristics that have implications for how the waste is processed (if necessary), stored, transported, disposed, and regulated. Along with the waste type, the geologic disposal concept will be used in conjunction with waste characteristics to form the technical basis to assess one (or more) potential Disposal Option(s) (i.e., a Disposal option is the combination of a disposal waste form with a disposal concept; SNL, 2014a). The assessment of a disposal option would identify the suite of issues related to that combination and rate the issues in a relative three tier system of (a) relatively easily addressed; (b) some analyses needed to address; (c) extensive analyses needed to address.

A final note - the purpose of this report is to evaluate potential pathways and strategies for the disposition of future Advanced Reactor Spent Nuclear Fuels and Advanced Reactor Waste Streams (i.e., these materials do not exist as yet), and as such, the detailed gap analyses and FEP analyses to be performed will be based on best available information for somewhat generic concepts of operations for the potential AR and bounding fuel cycles to assess AR SNF and other potential waste streams for storage, transportation, and disposal in generic (i.e., not site specific) repository concepts.

2 REGULATORY CONSIDERATIONS FOR ANALYZING AR SNF AND WASTE STREAMS

This section discusses the regulatory considerations for the back end of the nuclear fuel cycle (BENFC) associated with accident tolerant fuels (ATFs) and advanced spent nuclear fuels (AR SNF), including waste classification, reactor operations, storage, transportation, and disposal.

2.1 Waste Classification

The disposition options for discharged ATF and SNF from ARs and associated radioactive waste from reactor operations or fuel treatment will be based on their classification as SNF, HLW, TRU waste, low-level waste (LLW), or byproduct material. The definitions of these classifications are established by the DOE, Atomic Energy Act of 1954 and the 1982 NWPA (NWPA, 1983).

Spent nuclear fuel and HLW are defined in NWPA 1987 (NWPAA, 1987), Sec. 2 (23) and Sec. 2 (12)(A) and (B), respectively as follows:

> *Spent nuclear fuel is defined as fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing*
>
> *The term "high-level radioactive waste" means—*
>
> > *(A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and*
> >
> > *(B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.*

DOE has provided further clarification on HLW classification (DOE, 2019), as follows:

> DOE has revised its interpretation, as stated in its October 10, 2018, Federal Register Notice, after consideration of public comments. Based on those comments, DOE interprets the statutes to provide that a reprocessing waste may be determined to be non-HLW if the waste meets either of the following two criteria:
>
> I) does not exceed concentration limits for Class C low-level radioactive waste as set out in section 61.55 of title 10, Code of Federal Regulations, and meets the performance objectives of a disposal facility; or
>
> II) does not require disposal in a deep geologic repository and meets the performance objectives of a disposal facility as demonstrated through a performance assessment conducted in accordance with applicable requirements.

TRU waste is defined in 40 CFR 191.02(i) as follows:

> Transuranic radioactive waste, as used in this part, means waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes, with half-lives greater than twenty years, per gram of waste, except for: (1) High-level radioactive wastes; (2) wastes that the

> Department has determined, with the concurrence of the Administrator, do not need the degree of isolation required by this part; or (3) wastes that the Commission has approved for disposal on a case-by-case basis in accordance with 10 CFR Part 61.

LLW is defined in 10 CFR 61, *Licensing Requirements for Land Disposal of Radioactive Waste,* Section 61.2 as "radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in paragraphs (2), (3), and (4) of the definition of *Byproduct material* set forth in § 20.1003 of this chapter." The four classes of LLW are Class A, Class B, Class C, and Greater-Than-Class C (GTCC). 10 CFR 61 provides the waste form requirements and activity concentration for each class. The definition of byproduct material in 10 CFR 20, *Standards for Protection Against Radiation*, Section 1003 as follows:

> (1) Any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or using special nuclear material;
>
> (2) The tailings or wastes produced by the extraction or concentration of uranium or thorium from ore processed primarily for its source material content, including discrete surface wastes resulting from uranium solution extraction processes. Underground ore bodies depleted by these solution extraction operations do not constitute "byproduct material" within this definition;
>
> (3)(i) Any discrete source of radium-226 that is produced, extracted, or converted after extraction, before, on, or after August 8, 2005, for use for a commercial, medical, or research activity; or
>
> (ii) Any material that—
>
> (A) Has been made radioactive by use of a particle accelerator; and
>
> (B) Is produced, extracted, or converted after extraction, before, on, or after August 8, 2005, for use for a commercial, medical, or research activity; and
>
> (4) Any discrete source of naturally occurring radioactive material, other than source material, that—
>
> (i) The Commission, in consultation with the Administrator of the Environmental Protection Agency, the Secretary of Energy, the Secretary of Homeland Security, and the head of any other appropriate Federal agency, determines would pose a threat similar to the threat posed by a discrete source of radium-226 to the public health and safety or the common defense and security; and
>
> (ii) Before, on, or after August 8, 2005, is extracted or converted after extraction for use in a commercial, medical, or research activity.

The AR waste streams may impact storage, transportation, and disposal considerations for the other types of waste (other than SNF and HLW). It is recognized that substantial amounts of other types of wastes generated (e.g., LLW) have potential to substantially affect the overall total waste volume and potential

costs of decommissioning specific advanced reactor concepts, but the scope of this work is on the AR SNF (and other HLW waste streams, and possibly TRU) that would require deep geologic disposal.

Potential radioactive waste streams may also contain hazardous waste, warranting its classification as mixed waste. Hazardous waste is described in Per 40 CFR 261, *Identification and Listing of Hazardous Waste*. 40 CFR 261.30 provides a general description of hazardous wastes as follows:

> (a) A solid waste is a hazardous waste if it is listed in this subpart, unless it has been excluded from this list under §§ 260.20 and 260.22.
>
> (b) The Administrator will indicate his basis for listing the classes or types of wastes listed in this subpart by employing one or more of the following Hazard Codes:
>
> | | |
> |---|---|
> | Ignitable Waste | (I) |
> | Corrosive Waste | (C) |
> | Reactive Waste | (R) |
> | Toxicity Characteristic Waste | (E) |
> | Acute Hazardous Waste | (H) |
> | Toxic Waste | (T) |
>
> Appendix VII identifies the constituent which caused the Administrator to list the waste as a Toxicity Characteristic Waste (E) or Toxic Waste (T) in §§ 261.31 and 261.32.

Storage and disposal of mixed wastes are regulated by the Resource Conservation and Recovery Act (RCRA) (Resource, 1976) and the Atomic Energy Act of 1954, as amended. Waste that contains several different organic chemicals or non-organic lead, arsenic, barium, cadmium chromium, mercury, selenium, and/or silver is considered to be hazardous if those contaminants are present in concentrations that exceed specified regulatory levels (40 CFR 261.24).

2.2 Storage

Storage of ATF and SNF from ARs could be regulated by the NRC or regulated and operated by the DOE. SNF could be stored in one of the following configurations:

- Within the reactor building (e.g., in a reactor pool): Storage within the NRC-regulated reactor facilities is governed by 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*.
- At the reactor site in dry storage as an Independent Spent Fuel Storage Installation (ISFSI) governed by 10 CFR 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste* for NRC regulated facilities.
- Away from reactor sites: For example, GE Morris stores SNF in a pool governed by 10 CFR 72. This category is similar to storage on a reactor site that has been decommissioned.
- A DOE storage facility, which would be considered a Hazard Category 2 Nuclear Facility governed by 10 CFR 830, *Nuclear Safety Management*.

The NRC general storage requirements are provided in 10 CFR 72.122. The following is a list of key requirements:

> (b) Protection against environmental conditions and natural phenomena. (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI or MRS and to withstand postulated accidents.
>
> (2) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lighting, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions.
>
> (h) Confinement barriers and systems. (1) The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.
>
> (4) Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry spent fuel storage cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.
>
> (5) The high-level radioactive waste and reactor-related GTCC waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of part 20 limits. The package must be designed to confine the high-level radioactive waste for the duration of the license.
>
> (l) Retrievability. Storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related GTCC waste for further processing or disposal.

The NRC criticality safety requirements are provided in 10 CFR 72.124. The key requirements are:

> (a) Design for criticality safety. Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.

The NRC radiological protection requirements are provided in 10 CFR 72.126. The key requirements are:

> (a) Exposure control. Radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel.
>
> (d) Effluent control. The ISFSI or MRS must be designed to provide means to limit to levels as low as is reasonably achievable the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in § 72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in § 72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.

NUREG-2215, *Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities*, is the Standard Review Plan (SRP) that provides guidance to the NRC staff for reviewing safety analysis reports (SARs) for (1) a certificate of compliance for a dry storage system for use at a general license facility and (2) a specific license for a dry storage facility that is either an ISFSI or a monitored retrievable storage installation. NUREG-2215 does not apply to wet storage (e.g., GE-Morris). NUREG-2215 consolidates the guidance previously developed by the NRC in NUREG-1536, *Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility*, and NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*, as well as many Spent Fuel Storage and Transportation Interim Staff Guidance documents. Much of the guidance in NUREG-2215 pertains to storage of zircaloy-cladded uranium-oxide LWR SNF with a maximum enrichment of 5 wt.% U-235 and maximum burnup of 60 GWd/MTU. For SNF from ARs with significantly differing fuel matrix and cladding design, additional guidance in the form of Interim Staff Guidance documents or dedicated review plans will likely be developed.

The guidance in NUREG-2215 regarding storage packaging including baskets, neutron absorbers, confinement boundary (which is generally a welded canister, but can also be a bolted cask), and overpack/storage modules is directly relevant to ATF and may also be relevant to storage of SNF from ARs depending on the storage design and configurations.

NUREG-2215 refers to many guidance documents (NUREGs and RGs) for specific considerations (e.g., criticality control, dose calculations, and radiological monitoring). Applicability of this guidance to ATF and SNF from ARs would be dependent on the SNF and the storage system design. For example, RG 3.54, *Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation*, provides a methodology for calculating SNF heat generation rates as a function of burnup, specific power, decay time, and enrichment. This guide is applicable to fuels with a maximum initial enrichment 5 wt.% U-235 and burnup of 55 GWd/MTU 65 GWd/MTU for PWR and BWR fuels, respectively. Given that ATF and SNF from ARs would likely have design, enrichment and burnup characteristics differing than those covered by the guidance of RG 3.54, new revised or new guidance would have to be developed. Note that development of guidance is not a pre-requisite to submit a license application to the NRC.

Another example, RG 3.71, *Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores*, provides criticality safety guidance for operations with nuclear materials outside reactors, including storage and transportation under 10 CFR 72 and 10 CFR 71, respectively. This RG is generally agnostic to the fissionable material type and matrix and, therefore would apply to any ATF or SNF from ARs. For example, the ANSI/ANS-8 standards referenced in the RG are applicable to any system with sufficient amounts of fissionable materials to pose criticality potential (e.g., greater than 450 g Pu-239). However, several other guidance documents referenced in this RG that are specific to LWRs only. For example, NUREG/CR-7108, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions*, and NUREG/CR-7109, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (keff) Predictions,* are for LWR SNF (and in particular PWR SNF) burnup credit. Future R&D and modeling would likely facilitate developing similar guidance for ATF and SNF from ARs should burnup credit be used for storage, transportation, and disposal of these fuels.

Existing standards and guidance documents (in the form of RGs, NUREGs, NUREG/CRs, etc.) to address key safety performance requirements (e.g., confinement/containment, criticality, radiological protection) and relevant considerations (e.g., thermal management, structural design) would likely be reviewed to determine applicability and technical gaps important for the safe storage, transportation, and disposal of discharged ATF and SNF from ARs.

For DOE regulated and operated nuclear facilities, 10 CFR 830, Subpart B, Appendix A, Table 2 provides acceptable methodologies for preparing a documented safety analysis (DSA). DSAs for nonreactor nuclear facilities, including SNF storage, are prepared in accordance with DOE-STD-3009-2014, *Preparation of Nonreactor Nuclear Facility Documented Safety Analysis*. The contractor-prepared DSA is reviewed and approved by a DOE Field Office in accordance with DOE-STD-1104-2016, *Review and Approval of Nuclear Facility Safety Basis and Safety Design Basis Documents*. DOE Order 420.1C, *Facility Safety*, establishes facility design and programmatic safety requirements. Several DOE orders and standards are used to support a DSA for various technical areas (e.g., DOE-STD-1020-2016, *Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities*) and safety management programs. Of particular interest for storage facilities, which would be expected to have a relatively long design life, is DOE Order 433.1B, *Maintenance Management Program* DOE Order 433.1B requires the development and implementation of a nuclear maintenance management program (NMMP), a key requirement of which is *Aging Degradation and Technical Obsolesce* to evaluate aging-related degradation and obsolescence of safety systems.

2.3 Transportation

The NRC, DOE, DOT, and USPS regulate the safety of transportation of radioactive material. Regulations, regulatory guidance, orders, and standards issued by these agencies specify or provide guidance regarding functions and performance requirements for the transportation packaging (e.g., SNF shipping casks) and their approved contents to ensure the safety of the public and workers. The NRC transportation regulation are published in 10 CFR 71, *Packaging and Transportation of Radioactive Material*. DOE regulates certain non-commercial transportation of radioactive materials that originate from, or are used in, DOE and national defense facilities under DOE Order 460.1D, *Hazardous Materials Packaging and Transportation Safety* (DOE 2016).

The regulation of 10 CFR 71 provides specific safety performance functional requirements for packages including shielding, criticality, containment, and thermal limits for normal conditions of transport and hypothetical accident conditions (HACs). The acceptable design criteria for shipping casks and containment overpacks materials, design, fabrication, examination, and testing are provided in ASME Boiler and Pressure Vessel, Section III, Division 1, Subsection NB. The safety requirements in 10 CFR

71 pertinent to transportation of AR SNF include external radiation standards, requirements for Type B packages, and general requirements for fissile material packages. Demonstration of compliance is described in 10 CFR 71.41, which includes the following key requirement:

> (a) The effects on a package of the tests specified in § 71.71 ("Normal conditions of transport"), and the tests specified in § 71.73 ("Hypothetical accident conditions"), and § 71.61 ("Special requirements for Type B packages containing more than 10^5 A_2"), must be evaluated by subjecting a specimen or scale model to a specific test, or by another method of demonstration acceptable to the Commission, as appropriate for the particular feature being considered.

71.47 defines A_2 as *"...*the maximum activity of radioactive material, other than special form material, LSA, and SCO material, permitted in a Type A package. This value is either listed in Appendix A, Table A–1, of this part, or may be derived in accordance with the procedures prescribed in Appendix A of this part."

The external radiation standards are provided in 10 CFR 71.47, which includes the following key requirements:

> (a) Except as provided in paragraph (b) of this section, each package of radioactive materials offered for transportation must be designed and prepared for shipment so that under conditions normally incident to transportation the radiation level does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package, and the transport index does not exceed 10.
>
> (b) A package that exceeds the radiation level limits specified in paragraph (a) of this section must be transported by exclusive use shipment only, and the radiation levels for such shipment must not exceed the following during transportation:
>
> (1) 2 mSv/h (200 mrem/h) on the external surface of the package, unless the following conditions are met, in which case the limit is 10 mSv/h (1000 mrem/h):
>
> (i) The shipment is made in a closed transport vehicle;
>
> (ii) The package is secured within the vehicle so that its position remains fixed during transportation; and
>
> (iii) There are no loading or unloading operations between the beginning and end of the transportation;
>
> (2) 2 mSv/h (200 mrem/h) at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle; and
>
> (3) 0.1 mSv/h (10 mrem/h) at any point 2 meters (80 in) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and

> (4) 0.02 mSv/h (2 mrem/h) in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10 CFR 20.1502.

AR SNF would likely be transported in a Type B package. Specific requirements for Type B packages are provided in 10 CFR 71.51 as follows:

> (a) A Type B package, in addition to satisfying the requirements of §§ 71.41 through 71.47, must be designed, constructed, and prepared for shipment so that under the tests specified in:
>
> (1) Section 71.71 ("Normal conditions of transport"), there would be no loss or dispersal of radioactive contents--as demonstrated to a sensitivity of 10^{-6} A_2 per hour, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging; and
>
> (2) Section 71.73 ("Hypothetical accident conditions"), there would be no escape of krypton-85 exceeding 10 A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package.
>
> (b) Where mixtures of different radionuclides are present, the provisions of appendix A, paragraph IV of this part shall apply, except that for Krypton-85, an effective A_2 value equal to 10 A_2 may be used.
>
> (c) Compliance with the permitted activity release limits of paragraph (a) of this section may not depend on filters or on a mechanical cooling system.
>
> (d) For packages which contain radioactive contents with activity greater than 10^5 A_2, the requirements of § 71.61 must be met [i.e., must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than 1 hour without collapse, buckling, or inleakage of water.]

The general requirements for fissile material packages are provided in 10 CFR 71.55 as follows:

> (a) A package used for the shipment of fissile material must be designed and constructed in accordance with §§ 71.41 through 71.47. When required by the total amount of radioactive material, a package used for the shipment of fissile material must also be designed and constructed in accordance with § 71.51.
>
> (b) Except as provided in paragraph (c) or (g) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:
>
> (1) The most reactive credible configuration consistent with the chemical and physical form of the material;

(2) Moderation by water to the most reactive credible extent; and

(3) Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging.

(c) The Commission may approve exceptions to the requirements of paragraph (b) of this section if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak.

(d) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in § 71.71 ("Normal conditions of transport") --

(1) The contents would be subcritical;

(2) The geometric form of the package contents would not be substantially altered;

(3) There would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under § 71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material; and

(4) There will be no substantial reduction in the effectiveness of the packaging, including:

(i) No more than 5 percent reduction in the total effective volume of the packaging on which nuclear safety is assessed;

(ii) No more than 5 percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging; and

(iii) No occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (4 in) cube.

(e) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in § 71.73 ("Hypothetical accident conditions"), the package would be subcritical. For this determination, it must be assumed that:

(1) The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents;

(2) Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents; and

(3) There is full reflection by water on all sides, as close as is consistent with the damaged condition of the package.

NUREG-2216, *Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material*, provides guidance for reviewing applications for approval of package designs used for the transport of SNF and HLW under 10 CFR 71. Similar to NUREG-2215, applicability of this guidance to discharged ATF and SNF from ARs would be dependent on the SNF and the transportation packaging design. For SNF from ARs with significantly differing fuel matrix and cladding design, additional guidance or a dedicated review plan will likely be developed.

NUREGs, RGs, and standards references in NUREG-2216, similar to those referenced in NUREG-2215 for storage, may undergo revision and adaptation for discharged ATF and SNF from ARs. Identifying specific technical gaps in guidance for shielding, criticality, containment, and thermal will need to be performed. However, there are no apparent regulatory hurdles in the regulation for the safety storage of discharged ATF and SNF from ARs.

DOE has authority under the AEA of 1954, as amended, to regulate the transportation of radioactive materials. DOE safety requirements for proper packaging and transportation of offsite shipments and onsite transfers of hazardous materials, including radioactive materials, are provided in DOE O 460.1D, *Hazardous Material Packaging and Transportation Safety*. DOE O 460.1D, Section 4 states that "Department's packaging and transportation activities must be conducted in a manner that achieves an equivalent level of safety to that required by DOT and NRC for comparable commercial shipments". For offsite shipments, DOE O 460.1D, Section 4(a)(2) provides requirements for radioactive material packaging, including use of Type B or fissile materials certified packaging, use of DOT International Atomic Energy Agency (IAEA) certified packaging, application for NRC certified Type B or fissile material packaging, and application for departmental Type B or fissile material certified packaging. Per DOE O 460.1D, the application for departmental Type B or fissile materials certified packaging "must be supported by a Safety Analysis Report for Packaging (SARP). The SARP must adequately describe the proposed package in sufficient detail to identify the package accurately in accordance with 10 CFR 71.33 and provide a sufficient basis for evaluation of the package in accordance with 10 CFR 71.35. The application must also be supported by any other documentation needed to demonstrate the package meets the requirements of 10 CFR Part 71, Subparts D, E, F, G, and H, and any other applicable standards for certification." DOE O 460.1D describes the SARP required content as follows:

> A safety document (application) prepared by a Departmental element or contractors that describes and evaluates the proposed packaging and contents to be reviewed by appropriate DOE elements or NNSA elements, and serves as the basis for DOE CO or NNSA CO issuing a Certificate of Compliance. A SARP contains nine chapters: Chapter 1 General Information, Chapter 2 Structural Evaluation, Chapter 3 Thermal Evaluation, Chapter 4 Containment, Chapter 5 Shielding Evaluation, Chapter 6 Criticality Evaluation, Chapter 7 Package Operations, Chapter 8 Acceptance Tests and Maintenance Programs, and Chapter 9 Quality Assurance.

2.4 Disposal

This section focuses on disposal of AR SNF and potential HLW given the scope of this work. Disposal of SNF and HLW is currently subject to radiation protection standards and NRC regulations for geologic repositories, which are 10 CFR Part 60, *Disposal of High-Level Radioactive Wastes in Geologic Repositories*, and 10 CFR Part 63, *Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada*. 10 CFR 63 provides the regulatory requirements for disposing of SNF and HLW in a geologic repository at Yucca Mountain, Nevada, implementing the U.S. Environmental Protection Agency's 40 CFR 197 standard.

However, in light of the current situation in the US, where disposal is yet to be implemented, there exists a possibility that these regulations may be revised for other generic and/or specific sites. According to the recent National Academies of Sciences, Engineering, and Medicine report on Advanced Nuclear Reactors (2022): "Today, the United States has no viable regulatory basis for licensing a geologic repository for spent nuclear fuel, except at the Yucca Mountain site, because the regulatory framework is site specific…Any new repository site, even if sited in a willing community, will require licensing according to standards and regulations. The new standard and regulations may be site specific or generic." It is beyond the scope of this work and the above review of regulatory considerations to hypothesize how the standards and regulations might be altered.

2.5 Reactor Siting, Construction, Operation, and Decommissioning

Although reactor siting, construction, operation, and decommissioning are not part of the SFWST R&D scope, relevant regulatory structure and history are briefly discussed in this section for context. The primary NRC regulations for reactor siting, construction, operation, and decommissioning are 10 CFR Part 100, *Reactor Site Criteria*, 10 CFR Part 52, *Licenses, Certifications, and Approvals for Nuclear Power Plants*, and 10 CFR Part 50, *Domestic Licensing of Production and Utilization Facilities*. The primary review plan for SARs for nuclear power plants is NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,* with primary focus on LWRs. The review plan was appended in 2014 to address aspects specific to small modular reactors and an Appendix was added in 2017 for NuScale design-specific information.

Non-light water reactors similar to the reactor concepts under consideration were sited, licensed, and operated following the regulation documented in 10 CFR Parts 100 and 50. However, to address issues specific to ARs and incorporate the NRC's objective of migrating towards a risk-informed regulation, the NRC is developing a new proposed rule. 10 CFR 53, *Risk-Informed, technology-inclusive, Regulatory Framework for Advanced Reactors*, for optional use by applicants for new commercial advanced nuclear reactor licenses by December 31, 2027. The NRC staff presented its proposed plan for this rulemaking to the NRC Commission for approval in SECY-20-0032, which was approved in October 2020 (SECY-20-0032*, Rulemaking Plan on Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors*). The development schedule is provided in SRM-SECY-20-0032, Staff Requirements – SECY-20-0032 - *Rulemaking Plan on Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors*.

The NRC's predecessor, the Atomic Energy Commission, has some experience with non-LWR licensing that started with the first experimental breeder reactor in 1951 and ended by 1975. Since then, the NRC reviewed conceptual designs with varying detail, including the pebble bed modular reactor (PBMR) and General Electric-Hitachi PRISM in 2010. The NRC did not grant construction or operating licenses due to withdrawals by applicants and cancellations by DOE.

Three commercial non-LWRs were licensed by the Atomic Energy Commission and operated in the United States. These reactors were Fermi 1, a 200 megawatts thermal (MWt) sodium-cooled reactor near Newport, Michigan, Peach Bottom 1, a 115 MWt high temperature gas-cooled reactor (HTGR) near Delta, Pennsylvania, and Fort Saint Vrain, a 330 MWe HTGR near Denver, Colorado. DOE also built and operated several research and test non-LWRs, such as the Experimental Breeder Reactors (EBR-I and EBR-II) and the Molten-Salt Reactor Experiment.

The NRC was established in 1975 and reviewed its first construction permit application for the Clinch River Breeder Reactor Plant (a liquid sodium-cooled fast breeder demonstration reactor) in Oak Ridge, Tennessee and issued a safety evaluation report in 1983. Congress later voted to deny additional funding and the project was cancelled. The NRC conducted preapplication reviews of several other non-LWR

vendor designs, which were guided by NUREG-1226, *Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants* (NRC, 1988).

In its vision and strategy for licensing non-LWR reactors, the NRC acknowledges (NRC, 2016) that the "…agency needs to be effective and efficient as it conducts its safety, security, and environmental protection mission, without imposing unnecessary regulatory burden. This includes licensing reviews associated with fuel fabrication, storage, transportation and disposal." The proposed rule 10 CFR 53 does not provide any requirements or considerations related to the BENFC, including storage, transportation, and disposal. The NRC recently issued NUREG-2246, *Fuel Qualification for Advanced Reactors*, which provides a qualification assessment framework for AR designs. NUREG-2246 provides guidance related to neutronic performance (e.g., reactivity feedback), thermal-fluid performance (e.g., margin to critical heat flux limits), fuel mechanical performance, reactor core seismic behavior, fuel transportation, and storage. The scope of the assessment framework focuses on the identification and understanding of fuel life-limiting failure and degradation mechanisms due to in-reactor irradiation and environment. There are no discussions specifically aimed at storage and transportation, nor is disposal mentioned in the NUREG-2246.

2.6 Conclusions

The existing *regulations* for storage and transportation of SNF and HLW are generally agnostic to specific fuel or waste characteristics (with the exception of the information needed for classification). Therefore, updates or supplements to the existing regulations are not anticipated for AR SNF and ARWS, but would certainly be incorporated into these assessments if they occur.

The existing *regulatory guidance* is largely based on the low-enriched zircaloy clad uranium-oxide SNF. Significant updates or new guidance are suggested to explicitly address BENFC implementation/analyses for AR SNF and ARWS. The primary technical concepts for consideration in updating regulatory guidance relate to the following areas:

- Storage confinement
- Transportation containment
- Retrievability during storage
- Geometry control during transportation
- Thermal and structural performance criteria during loading, storage, and transportation operations
- Burnup credit and criticality control
- Source term development

The guidance is expected to be highly dependent on the diversity of fuel designs (e.g., different fuel components) and reactor irradiation characteristics. Therefore, some of the guidance (e.g., burnup credit) would likely not be available until significant reactor irradiation experience has been gained.

3 SURVEY OF ADVANCED FUELS AND ADVANCED REACTOR WASTE STREAMS

Advanced nuclear reactors are being driven by private and venture capital investments in many of the generation-IV nuclear reactor designs. These designs have differing coolants, fuels, and configurations compared to traditional large light water reactors (LWRs). This chapter also discusses accident tolerant fuels (ATFs), which are intended to be used in existing LWRs. These advanced technologies have the goal of making nuclear reactors safer, more affordable, and more efficient. Many of the advanced reactor fuels (ARFs) allow for higher burnup, which increases the lifetime of the fuel. This would allow vendors to increase the time between refueling and buying new fuel, thus lowering costs. ATF technologies intend to make the fuel safer in the case of an accident. Qualities like higher thermal efficiency and lowering the amount of gaseous release are desired to minimize or prevent exposure/release in the event of an accident. This section discusses proposed advanced reactor fuels (Section 3.1), ATFs (Section 3.2), and the AR concepts that use these fuels (Section 3.3).

3.1 Advanced Reactor Fuels

3.1.1 TRISO Fuels

TRi-structural ISOtropic (TRISO) pellet fuels were designed originally for high temperature gas reactors and are tolerant to very high temperature operation and transient conditions. These fuels have been proposed for pebble bed reactors (PBRs) and prismatic block high temperature gas reactors. PBR use of TRISO fuel could be implemented either in an HTGR or in a molten-salt cooled reactor. Reactors using TRISO fuel may employ low-enriched uranium (LEU) or high assay low-enriched uranium (HALEU) fuels. The TRISO fuels were originally designed for once-through operation—reprocessing is possible but challenging given the effort to remove the fuel from each pellet.

TRISO fuel particles consist of a fuel kernel surrounded by carbon-based layers (e.g., Sassani et al., 2018; Sassani and Gelbard, 2019). The fuel kernel is typically UO_2 or a mixture of UO_2 and UC_2, referred to as UCO. A UO_2-based kernel is the more typical design. The kernel has a diameter of around 500 microns for both pebble type fuels and fertile prismatic fuel. It is roughly 350 microns for fissile prismatic kernels. The layers of the fuel can be seen in Figure 3-1. The layer covering the kernel is the Inner PyroCarbon (IPyC) layer. This is followed by the silicon carbide (SiC) layer and the Outer PyroCarbon layer (OPyC).

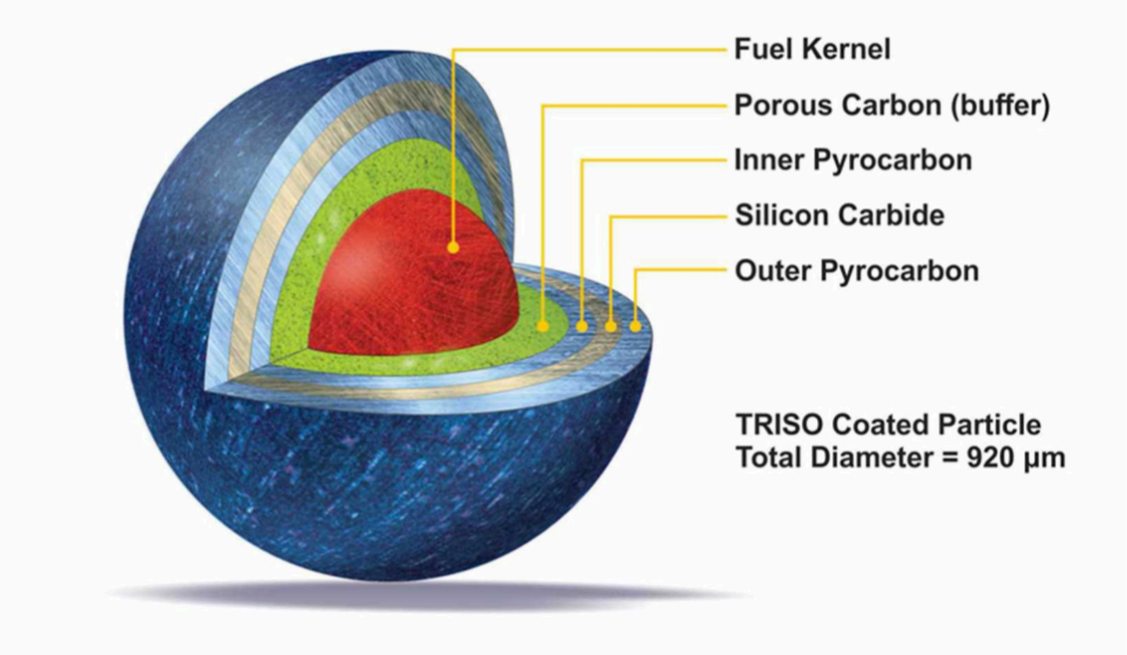

Figure 3-1. Typical Layers in a TRISO Fuel Particle (Sassani et al., 2018)

TRISO particles are the smallest subunit of the TRISO fuel and a single fuel kernel/particle with multiple layers is represented in Figure 3-1 The relative sizes of coated particles, fuel compacts, and graphite fuel elements are pictured in Figure 3-2. For a Prismatic Block Reactor (PMR), the coated particles (~ 1mm in diameter) are encapsulated in graphitic fuel compacts that are tens of mm in diameter and ~10 cm length, which can be arrayed in a prismatic graphite fuel element, as shown in Figure 3-2.

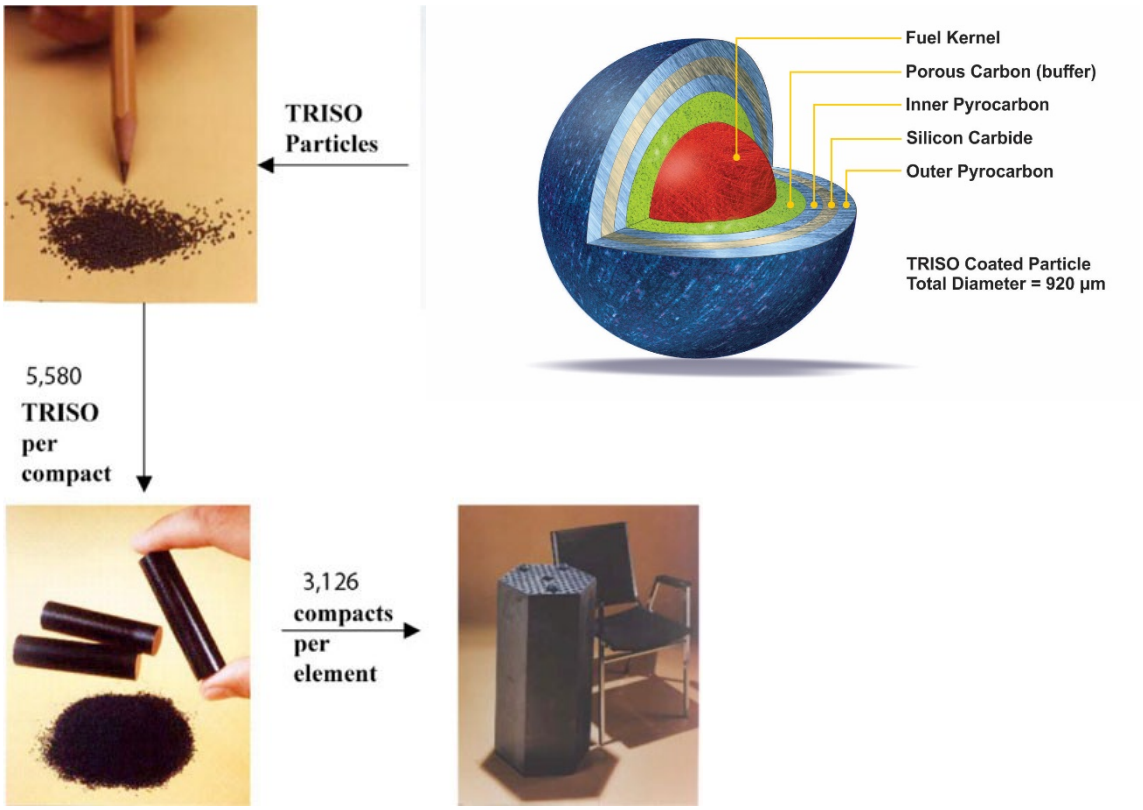

Figure 3-2. TRISO coated particles (left top) are encapsulated in graphitic compacts (left bottom), which are inserted into prismatic graphite fuel elements (bottom right) (from Sassani et al., 2018 Figure 3-21 that used images from Sterbentz et al., 2004 and values from Van den Akker and Ahn 2013).

In a pebble bed reactor (PBR), freely moving spherical graphite fuel units (i.e., pebbles, instead of compacts in prismatic fuel elements) are used to contain the TRISO particles. These pebbles have a diameter of on the order of tens of millimeters and are a mix of graphite and TRISO coated particles, typically with a 5mm dense graphite coating surrounding a coated particle/graphite matrix, as shown in Figure 3-3.

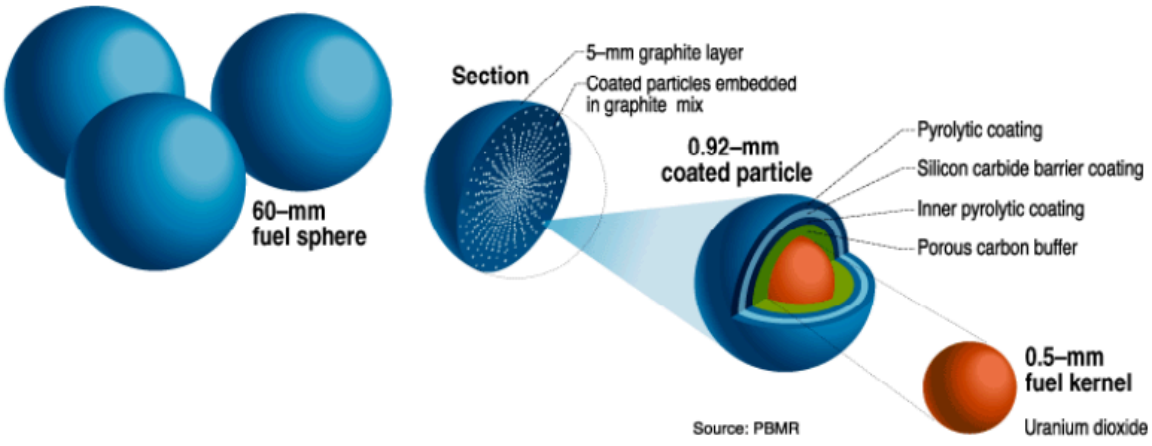

Figure 3-3. TRISO Fuel Pebble Schematic (Windes, et al., 2014)

Both types of TRISO fuel units (compacts and pebbles) will contain thousands of TRISO particles per unit. Typically, TRISO fuels are designed to have burnups in the range of 20-25% Fissions per Initial Metal Atom (FIMA) (NRC, 2004). Specifications on the MIT pebble bed reactor and power specifications from Kairos Power are shown in Table 3-1 (Kadak, 2005) and Table 3-2 (Kairos Power, 2022), respectively, below.

Table 3-1 German Specifications for Pebble Reactor and Fuel Element (Kadak, 2005).

Outer Diameter	60.0 mm
Fuel-free shell thickness	5.0 mm
Uranium mass/fuel pebble	7.0 g
U-235 enrichment	8.0 %
Power Output	250 MW - 120 MWe
Target Thermal Efficiency	45%
Helium Pressure	80 bar
Helium Entry/Exit Temp	520°C/900°C
Core Height	10.0 m
Core Diameter	3.5 m
Pressure Vessel Height	16 m
Pressure Vessel Radius	5.6 m
Number of Fuel Pebbles	360,000
Mean Power Density	3.54 MW/m^2

Table 3-2 Kairos Power Reactor Specifications (Kairos Power, 2022).

Power Output	140 MWE
Net Efficiency	45%
Main/Reheat Temp	585°C/ 585°C
Main Steam Pressure	19 MPA
Reactor Outlet Temp	650°C
Reactor Operating Pressure	Near Atmospheric
Reactor Structural Material	316 SS
Fuel Enrichment	19.75%
Intermediate Salt	"Solar"/Nitrate Salt
Refueling Type	Online
Deployment Configurations	Single / Multi-module

Additional information on TRISO fuel and HTGRs can be found Appendix A.

3.1.2 Metallic Fuels

Metallic fuels were originally developed for sodium-cooled fast reactors, are generally composed of uranium or plutonium metal alloy as the fuel, and have excellent heat transfer properties. Fast reactors allow for higher-Z actinides to be fissioned, thus preventing a buildup of higher-Z actinides in the fuel cycle. As such, reprocessing was originally envisioned for fast reactor fuel in order to recycle all Pu and minor actinides. However, once-through disposal options should also be considered.

Many metallic fuels use sodium-bonded technology. That is, sodium is placed in the fuel rod to facilitate heat transfer from the metallic fuel to the fuel cladding.

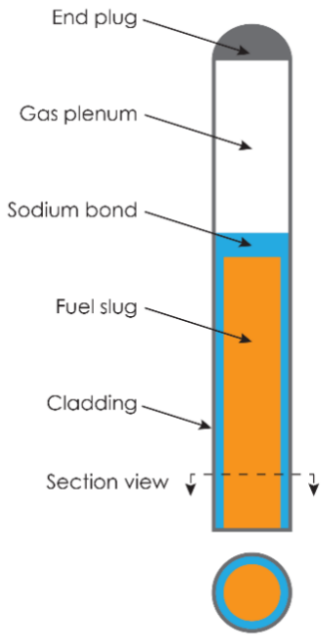

Figure 3-4. Metallic fuel diagram containing sodium bond (FRWG, 2018)

Disposal options for this type of fuel will likely entail treatment for removal or deactivation of the sodium (INL, 2007) because metallic sodium is highly reactive in both air and water. In 2000, the DOE determined that direct disposal of sodium-bonded SNF would not meet DOE or NRC repository acceptance criteria, mainly due to the potential for pyrophoric reaction of metallic sodium with water with which it may come into contact (DOE, 2000). The National Academy of Sciences also concurs that sodium-bonded spent fuel is not suitable for direct disposal because of the highly reactive and pyrophoric characteristics of sodium (NASEM, 2022). Advanced metallic fuels that do not contain sodium bonded in the fuel rod have been proposed and are being developed (NASEM, 2022). These fuels would use helium in the core as a heat-transfer fluid, rather than metallic sodium (TerraPower, 2022a). Further FEP evaluations for disposal of metallic fuels will include assessment of reactivity/pyrophoricity in generic disposal systems.

3.1.3 Molten Salt Fuels

Molten salt reactors (MSR) have also seen a resurgence in interest (see Section 3.3.3 for discussion of commercial concepts). Molten salt reactors may have either fluoride-based salt fuel, for example the Molten-Salt Reactor Experiment (MSRE) at Oak Ridge National Laboratory (ORNL), or chloride-based molten salt fuel. In addition to these MSR fuel options - the coolant molten salt for some TRISO pebble-bed reactors (as discussed above) may end up contaminated with some fuel materials such that they may be subject to deep geologic disposal and therefore also -discussed here because of similar chemical form. The MSR designs span (1) those fueled with a molten salt "core" that is replaced every 7-8 years with only very limited on-site treatment, to (2) those with a molten salt "core" that is continuously treated for fission product removal on site. Liquid fueled MSRs can have either thorium or uranium as the fuel and can operate in the fast or thermal spectrum. Regardless, MSRs will result in some molten salt waste stream likely requiring disposal or treatment. A benefit of MSRs is onsite treatment to remove fission products from the molten salt fuel. It is not currently clear (a) how much and (b) of what composition, such treated/removed materials may be. Such material when specifically defined may be classified as HLW and would be subject to evaluation for deep geologic disposal as a separate MSR waste stream but has not yet been included in this report.

Limited data are available for MSR spent fuel. Most data are from the Molten-Salt Reactor Experiment (MSRE) at Oak Ridge National Laboratory (ORNL) for that fluoride-based molten salt fuel. However, some preliminary work has been done on the properties of various salts and their suitability as a coolant for MSRs. Table 3-3 shows the physical properties and Table 3-4 shows the thermal and flow properties of many of the salts being considered.

Table 3-3 Molten salt physical properties (Sohal *et al.*, 2010).

Coolant	Mol% (Wt%)	Formula Weight (g/mol)	Freezing/ Melting (K)	Density (kg/m^3)
LiF-NaF-KF	46.5-11.5-42 (29-12-59)	41.3	727	2020
LiF-BeF2	67-33 (53-47)	33.0	733	1940
KCl-MgCl2	68-32 (62-38)	81.44	708	1664
NaNO3-NaNO2-KNO3 @400C	7-49-44	1.77-1.98	415	1790

	(7-40-53)			
NaNO3-KNO3 @400C	48-52	93.37	495	1840
LiF-NaF-BeF2	--	38.9	588	2000
LiF-NaF-RbF	--	67.7	708	2690
NaF-BeF2	--	44.1	613	2010

Table 3-4 Molten salt thermal and flow properties (Sohal *et al.*, 2010).

Coolant	Specific heat capacity (j/kgK)	Viscosity (Pa-s)	Thermal conductivity (W/mK)	Prandtl No.
LiF-NaF-KF	1882.8	0.0029	0.92	5.938
LiF-BeF2	2414.17	0.0056	1.0	13.525
KCl-MgCl2	1158.97	0.0014	0.40	4.0
NaNO3-NaNO2-KNO3 @400C	1560	0.0013-0.0016	0.51-0.605	--
NaNO3-KNO3 @400C	2660.19	0.0017	0.55	8.222
LiF-NaF-BeF2	2045.98	0.005	0.97	10.551
LiF-NaF-RbF	987.42	0.0026	0.62	4.14
NaF-BeF2	2175.68	0.007	0.87	17.513

3.2 Accident Tolerant Fuels

Accident tolerant fuels can cover a range of fuels including new designs for existing LWRs, examined in this report to assess some of the differentiating characteristics (relative to traditional LWR fuel) that are relevant to understanding characteristics of AR fuels. The properties of ATFs are designed to improve the severe accident response time and performance of fuel. They are split into two categories, fuel technologies (or pellet technologies) and cladding technologies, which are explored in the following sections. Below, Table 3-5 lists the licensees that have inserted lead test assemblies with ATF technologies.

Table 3-5. U.S. Power Reactors Licensees that have inserted Lead Test Assemblies with ATF Technologies (NRC, 2022a)

Technology	Plant	Licensee	Fuel Vendor	Date Inserted
Doped Pellets	Byron	Exelon	Westinghouse	Spring 2019
Doped Pellets	Vogtle	Southern Nuclear	Framatome	Spring 2019
Doped Pellets	ANO	Entergy	Framatome	Fall 2019
Uranium Silicide Pellets	Byron	Exelon	Westinghouse	Spring 2019
Chromium-coated cladding	Hatch	Southern Nuclear	GNF	Spring 2018
Chromium-coated cladding	Byron	Exelon	Westinghouse	Spring 2019

Chromium-coated cladding	Vogtle	Southern Nuclear	Framatome	Spring 2019
Chromium-coated cladding	ANO	Entergy	Framatome	Fall 2019
Chromium-coated cladding	Clinton	Exelon	GNF	Fall 2019
FeCrAl Cladding	Hatch	Southern Nuclear	GNF	Spring 2019
FeCrAl Cladding	Clinton	Exelon	GNF	Fall 2019

3.2.1 Fuel Technologies

Doped pellets are a technology being explored by Westinghouse. The dopants change the physical properties of the pellet (e.g., reduced rigidity or increase ceramic grain size). Westinghouse's ADOPT™ fuel is doped low-enriched uranium (LEU) (Westinghouse, 2022a). It is doped with Cr_2O_3 and Al_2O_3. They claim it has greater uranium efficiency, enables higher burnup, lowers fission gas release, has a higher thermal stability, increases pellet clad interactions, and enhances corrosion and washout resistance. Testing was done on their ADOPT™ pellets (Hallman Jr., 2020); however, most of the relevant information is proprietary and has been removed from the document. It was seen that the fission gas release of the doped pellets was around 17% vs 30% for their normal UO_2 fuel. Their pellets were enriched to 4.2% and were tested in a BWR with a ramp test. This fuel technology is part of their EnCore™ fuel. The ADOPT™ pellet is surrounded by their chromium cladding and arrived at ORNL in August 2021 to be irradiated for one year.

Other technologies include increased enrichment and higher burnup. Increased enrichment will increase operational flexibility, minimize assemblies to be purchased, and achieve higher burn up. Higher burnup allows for increased time between refueling and minimizing assemblies to be purchased (NRC, 2022b).

Westinghouse is also working on developing a uranium nitride (UN) fuel, which replaces previous efforts to develop uranium silicide as a UO_2 replacement (NRC, 2022b). Like uranium silicide fuel (which were found to be non-viable after lead test assembly testing), UN fuels are 30-40% denser than their oxide counter parts, which allows for higher burnup and longer fuel cycles. Nitrides are also known to be better thermal conductors than oxides. Nitrides fuels will also dissolve in nitric acid more readily, making it more easily reprocessed. However, nitride fuels are also likely to oxidize in superheated steam and readily reacts with the oxygen in the air, making it more difficult to handle than its oxide counterparts (Ekberg, et al., 2018). Table 3-6 shows the difference between UO_2 fuel and UN fuel (Ma, 1983).

Table 3-6. Properties of UO_2 vs UN Fuel

	UO_2	**UN**
Theoretical Density [g/cm^3]	10.96	14.32
Heavy Metal Density [g/cm^3]	9.67	13.52
Specific Heat [J/kg-K] @ 1000°C	330	230
Melting Point [°C]	~2800	~2700
Thermal Conductivity [W/m-K] @ 200°C	7.19	12
Thermal Conductivity [W/m-K] @ 1000°C	3.35	20

Linear Thermal Expansion Coeff [10^{-6}/K] @ 1000°C	11.5	9.2

3.2.2 Cladding Technologies

A variety of cladding technologies are being considered for ATFs. These include FeCrAl, coated, zirconium-chromium alloy, and silicon carbide. FeCrAl cladding could improve high-temperature oxidation, strength at normal operation conditions and high temperature accident conditions, and corrosion performance at normal operation (NRC, 2022a). A study was performed on both zirconium-chromium alloy cladding and silicon carbide cladding using the fuel performance model, BISON (Wagih et al., 2018). They were tested in steady state pressurized water reactor (PWR) conditions, ramp conditions, and loss-of-coolant accident (LOCA) conditions. Images of the different claddings are shown below in Figure 3-5. The test rod specifications are found Table 3-7. The results concluded that both show promise and performed similarly to the Zr4 cladding in all scenarios. The SiC fuel under LOCA conditions was not predicted to burst. Cr-coated Zr4 cladding show an order of magnitude of improvement in oxidation resistance, and the fuels should be tested experimentally (Wagih et al., 2018)..

The presence of additional Cr in the cladding, compared to typical LWR cladding, raises the question of whether the SNF would be considered hazardous under the RCRA toxicity characteristic, as defined in 40 CFR 261.24, and would thus be subject to regulation under RCRA. The EPA commissioned an evaluation of SNF as mixed waste (EPA, 1990) and determined that the concentration of Cr, which was the only contaminant of concern, was well below the regulatory level. Therefore, SNF was not considered to be mixed waste. This evaluation might need to be repeated for ATF that contains additional Cr, compared to typical fuel.

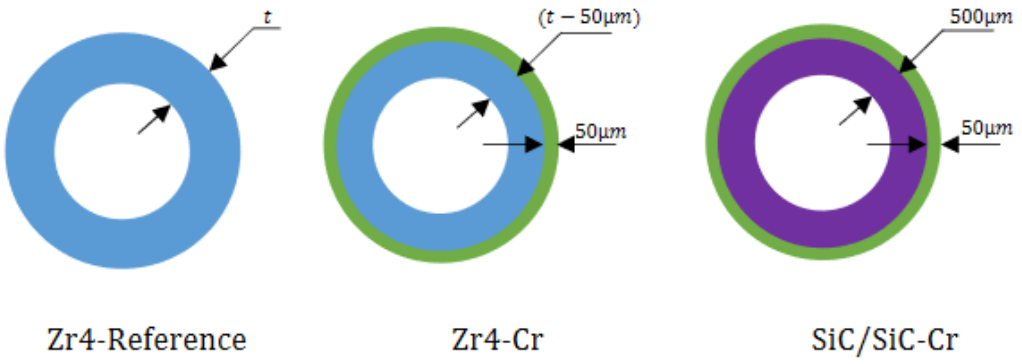

Figure 3-5. Chromium-coated Zr4 and SiC claddings.

Table 3-7. Steady state test rod specifications (Wagih et al., 2018).

	Steady State	Ramp Test	LOCA Test
Overall Length		52.24 cm	380 cm
Fuel Stack Height	15 cm	43.295 cm	366 cm
Nominal Plenum Height	0.75348 cm	8.944 cm	20 cm
Number of pellets per rod	15	32	1
Fill gas composition	He	He	He
Fill gas pressure	4.2 MPa	2.6 MPa	4.62 MPa
Fuel enrichment	5%	4.5%	2.93%
Density	95%	95.73%	95%
Outer Diameter	8.1919 mm	8.192 mm	8.26 mm
Gain Diameter	5 µm	10 µm	7.8 µm
Pellet Geometry	Dished	Dished	Smeared
Dish Diameter	21.2 mm	6 mm	
Dish depth	0.287 mm	0.32 mm	
Chamfer width	0.5 mm		
Chamfer depth	0.16 mm		
Cladding Outer Diameter	9.5 mm	9.5 mm	9.63 mm
Cladding Inner Diameter	8.357 mm	8.36 mm	8.41 mm
Cladding Wall Thickness	0.5715 mm	0.57 mm	0.61 mm
Coating thickness (for both Zr4-Cr and SiC/SiC-Cr)	0.005 mm	0.005mm	0.005mm (both), 0.0025mm (Zr4-Cr)

3.3 Overview of Commercial Advanced Reactor Concepts

Reactor vendors are pursuing advanced technologies, and many hope to be in operation in the mid-2020s to mid-2030s. They are briefly discussed below and are categorized by fuel technology.

3.3.1 TRISO Fueled Advanced Reactors

Most advanced reactor vendors are moving towards TRISO fuel. X-energy, BWXT, Framatome, Kairos, and Westinghouse all will use UCO fuel in a TRISO particle. X-energy, BWXT, and Framatome are all HTGRs that use Helium as the coolant. X-energy has two reactors, Xe-100 and Xe-Mobile, where the Xe-Mobile uses the same technologies as Xe-100 but is smaller to serve remote locations and microgrids and only uses High Assay Low-Enriched Uranium (HALEU). Xe-100 will produce 80MWe, but four can be combined for a total of 320MWe, compared to the 1-5MWe produced by Xe-Mobile. It will offer online refueling and can use HALEU or LEU. Both reactors are scheduled for deployment in 2027.

BWXT's Advanced Nuclear Reactor (BANR) uses HALEU and will produce 17MWe at maximum efficiency. BANR is scheduled for testing at Idaho National Laboratories (INL) in 2024. Framatome's Steam Cycle HTGR (SC-HTGR), uses helium as the primary coolant and will have a max output of

272MWe. The SC_HTGR will use HALEU for fuel and has a prismatic block annular configuration. It is set to deploy by 2027 (BWXT, 2022; Gateway for Accelerated Innovation in Nuclear, 2021; X-Energy, 2022a; X-Energy, 2022b).

Kairos Power's Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR) uses Fluorine-Lithium-Beryllium (FLiBe) molten salt as it's coolant. The plant will have two salt loops, FLiBe being the primary and Nitrate salt being the secondary. It uses HALEU and will produce 140MWe. Kairos plans to begin construction in 2023 with operation beginning in 2026 (Kairos Power, 2022).

Westinghouse's eVinci microreactor is modelled after EBR-II and uses sodium as the coolant and heat pipes to move the heat away from the core. It will produce 5MWe and uses HALEU. eVinci is set to be online by March 2026 (Westinghouse, 2022b; BWXT, 2022; Gateway for Accelerated Innovation in Nuclear, 2021; X-Energy, 2022a; X-Energy, 2022b).

Unlike the other reactors named, Ultra Safe Nuclear Corporation's Micro Modular Reactor (MMR) will use traditional uranium oxide fuel in TRISO. It will operate on HALEU and will produce 5-10MWe. Like other reactors mentioned, multiple reactors can be used together to increase power as needed. It uses helium as the primary coolant, and current plans are to bury the reactor. It is scheduled for release in 2026 (Ultra Safe Nuclear Corporation, 2022).

3.3.2 Metallic Fueled Advanced Reactors

A few of the reactors considering metallic fuel are Oklo's Aurora, TerraPower and GE-Hitachi's Natrium, and ARC Clean Energy's ARC-100, and they indicate an intent to employ a sodium-bonded metallic uranium alloy fuel, at least initially. The Natrium system plans to transition from a sodium-bonded metallic fuel to an advanced metallic without sodium bonding to the cladding (NASEM, 2022; TerraPower, 2022a). These reactors plan to use HALEU for their fuel. Both Natrium and ARC-100 are using liquid sodium as the coolant while Aurora does not have a flowing coolant in the core, but instead relies on heat pipes to dissipate the heat. These reactors vary greatly in size. Aurora is planned to produce 1.5MWe, while the ARC-100 is sized to 100MWe, and Natrium is set to be the largest at 345MWe. Aurora applied for license to the NRC, and as of January 6th, 2022, was denied its licenses application for failure to supply information. Both Natrium and ARC-100 have an implementation date of the late 2020s, with ARC-100 specifically hoping to have online in Canada by that time (Gateway for Accelerated Innovation in Nuclear, 2021; TerraPower, 2022b).

3.3.3 Molten Salt Fueled Advanced Reactors

For molten salt reactors, the fuel options are either liquid fueled or TRISO fueled. TerraPower and Southern Company's joint Molten Chloride Fast Reactor (MCFR) and Terrestrial Energy's Integral Molten Salt Reactor (IMSR-400) are two emerging designs. The MCFR will use HALEU, operate in the fast spectrum, and produce 800MWe. The IMSR-400 will use LEU, operate in two thermal spectrums, and produce just half of the MCFR at 400MWe. The MCFR will start Integrated Effects Tests (IET) to start in Washington state in 2022. The IMSR-400 boasts a replaceable core that contains a graphite moderator, has three salt loops, and is set to be deployed by the late 2020s (TerraPower, 2022c; Terrestrial Energy, 2022).

Moltex Energy's Stable Salt Reactor (SSR-U) has a different design. It will contain fueled salt in vented tubes, and a different molten salt will be the primary coolant. It uses 6% enriched uranium and is scheduled for refueling every 16-20 years. The primary and secondary coolants are fluoride eutectic salts that do not contain either lithium or beryllium to prevent tritium production. Moltex is also planning to build a waste burning reactor (SSR-W) that will produce 300MWe. The SSR-W has cleared the first

phase of regulatory review in Canada. The SSR-W is planned to come online in New Brunswick in the 2030s (Moltex Clean Energy, 2022).

3.3.4 Reactors using Accident Tolerant Fuel Cladding

General Atomics (GA) is building two reactors that use Accident Tolerant Fuel (ATF) that they are developing that is a silicon-carbide composite called SiGA®. GA claims SiGA is meltdown proof as the composite can withstand more than twice the temperature of the metal components. The main difference between the two reactors is the size. The Fast Modular Reactor (FMR) produces 50MWe while the Energy Multiplier Module (EM^2) produce 265MWe per unit, with an option of combining up to 4 units for 1GWe. They are both High Temperature Gas Reactors (HTGR) and use helium as the coolant. Both reactors will be 53% efficient, allowing the FMR to go 9 years without refueling and the EM^2 to go 30 years without refueling. The FMR will use HALEU, while the EM^2 will use Low Enriched Uranium (LEU), but it can also use spent nuclear fuel. The FMR is set to be deployed first in the early to mid-2030s with the EM^2 to follow shortly after. (General Atomics, n.d.; General Atomics, 2021)

3.3.5 Advanced LWR Fueled - Small Modular Reactors and Microreactors

Advanced light water reactor designs plan to come online as early as 2028 (GE Hitachi, 2023). While these designs use the same technology as traditional LWR, they are small modular reactors (SMRs), which, like the name suggests, are smaller in design allowing for deployment in remote areas. All three front runners (Holtec's SMR-160, GE-Hitachi's BWRX-300, and NuScale) plan to use LEU fuel for their designs (Holtec, 2023; IAEA 2019a; IAEA, 2019b; and NuScale, 2023). Additionally, all boast enhanced safety features. NuScale's reactors are designed to be below grade (NuScale, 2022), while both Holtec and GE-Hitachi include natural circulation in their design, so the reactors are not reliant on pumps for cooling (Holtec 2023, GE Hitachi 2023). Some SMRs may follow similar disposal paths as current LWR designs. However, some SMR or microreactor designs have mentioned fully swapping out the reactors once the core was depleted. Transporting reactors full of spent nuclear fuel is not within the scope of this report, and therefore is not discussed here. For more discussion on the regulation of transportation of whole reactors and from sites, please see Nuclear Waste Attributes of SMRs Scheduled for Near-Term Deployment by Kim et al. (2022) or Chapters 2 and 7 in Laying the Foundation for New and Advanced Nuclear Reactors in the United States by the National Academy of Sciences (NASEM, 2023).

3.4 Waste Characteristics of Advanced Reactor Fuel and Advanced Reactor Waste Streams

Existing, developing, and conceptual advanced fuels, such as ATF- for existing fuel cycles and AR fuels for potential new fuel cycles, exhibit a broad spectrum of characteristics, which are crucial to evaluating disposal performance of the waste at the back end of the nuclear fuel cycle (BENFC). In comparison to standard light-water reactors (LWR), such as PWR and the BWR, which have well-defined characteristics, disposition strategies for most spent AFs and ARWSs are not currently as well-defined.

This stems in part from the fact that most commercial ARs are not expected to be available before *ca*. 2030, as well as from the protection of proprietary information on innovative AFs under development by various advanced nuclear reactor and fuel companies. The disposition solutions may ultimately differ depending on AF characteristics and repository waste acceptance criteria, as well as numerous other criteria related to the reduction of the overall impact of the waste produced (e.g., minimization of waste radiotoxicity and thermal load, minimization of the total amount of waste produced during AF manufacturing, reactor operation and decommissioning, minimization of natural resource utilization, or other important criteria such as waste form performance, proliferation resistance, and costs associated with AF and waste forms production (IAEA, 2019c). Although valuable operational and

decommissioning experience was gained from previous experimental fast reactors (IAEA, 2004), waste generated from commercial ARs might also differ from test reactors owing to different operation conditions between commercial and experimental ARs. In addition, options for processing and disposition of waste may have evolved since these seminal attempts at developing and testing concepts for ARs.

This section discusses desirable generic waste characteristics relevant to repository performance in existing generic repository concepts (e.g., Swift and Sassani, 2020; Sassani et al., 2022). The focus here is on waste characteristics that are important to disposal, the reason for which are as follows: (1) the time period that disposal must account for is several orders of magnitude longer than the time period that must be considered for storage or transportation and (2) generally speaking - and relative to disposal - storage and transportation will be mostly agnostic to many waste characteristics except for burn-up and other characteristics of the fuel that will remain largely unknown until more experience is gained by irradiating the fuel. While disposal performance may have more dependence on waste form characteristics such as, durability and thermal output, it is important to note that there is a complex interplay between waste form durability, geosphere transport time, and the characteristics (e.g., solubility and sorptivity) of the radionuclides that dominate the estimated dose calculation (Swift and Sassani, 2020; Sassani et al., 2022). All three of these parameters and their effect of estimated dose would depend on the repository host lithology and design concept.

3.4.1 Overview of Waste Characteristics

Although many different possible combinations of factors and metrics can be utilized to evaluate disposal system performance for spent fuels produced from various fuel cycles, this section summarizes – in no specific order of importance – some of the desirable, generic waste characteristics in terms of radionuclide inventory, thermal output, chemical reactivity, physical aspect and dimensions, packaging, and safeguards and security needed for handling and disposing of the prospective spent AFs and possible waste forms in generic repository concepts.

3.4.1.1 Radionuclide Inventory

The radionuclide inventory, consisting of actinides and fission or activation products, is one of the crucial characteristics of spent fuels and waste forms, since it corresponds to the primary hazard at the BENFC, which can be made safe by deep geologic disposal. Of particular concern for disposal strategies are mobile, long-lived fission products and activation products, such as ^{99}Tc, ^{129}I and ^{36}Cl (with half-life > 10^5 year), which can significantly affect the long-term dose in any repository. It is worth noting that while the radiotoxicity of the waste is strongly tied to its actinide content, actinide mobility under typical repository conditions is relatively slow with respect to the long-lived fission products and activation products. In this context, the quantity of fission products in the waste/spent fuel which increases with spent fuel burnup can become important, especially since many of the AFs are expected to reach higher burnup levels under normal operation conditions compared to traditional fuels utilized in LWRs. Some of the short-lived fission products in the radionuclide inventory are also important in the context of thermal management of the waste: ^{90}Sr and ^{137}Cs (with half-life of ~30 years) and their daughter isotopes are of particular interest, especially if actinides have been extracted from the spent fuel or waste by partial reprocessing. Moreover, wastes containing a high level of fissionable material are also noteworthy among the radionuclide inventory owing to their potential post-closure criticality and with regards to safeguards and security; this is expected to be the case for spent HALEU with ^{235}U enrichment up to ~20% for smaller design ARs or for spent TRISO fuel with high enrichment. Detailed fuel composition information, including fissile material, actinides, and fission products important to criticality (i.e., stable or long-lived with appreciable neutron absorption cross sections) are needed for criticality calculations for the existing repository concepts; in salt repositories, criticality calculations might be limited only to higher-enriched fuels (i.e., greater than 5 wt.% U-235), since the likelihood of a criticality event for low-enriched fuel is practically

zero in a salt repository due to the high concentration of chlorine, which provides a relatively significant negative reactivity (Rechard, 1993; Rechard *et al*., 1996). For higher enriched fuels, additional criticality control may be needed. Finally, other potentially problematic radionuclides, such as tritium (^{3}H) and ^{14}C, may be challenging during handling and processing of waste in preparation for disposal.

3.4.1.2 Thermal Characteristics

The thermal output of the spent fuels and waste forms is a central factor in the design and management of deep geologic repositories, as well as in handling of the waste prior to disposal. The radionuclides typically responsible for most of the thermal output are ^{90}Sr, ^{90}Y, ^{137}Cs, ^{137m}Ba, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, and ^{244}Cm (Gauld & Murphy, 2010). As mentioned above, the thermal output of the waste during the initial few hundred years is dominated by the heat produced by ^{90}Sr and ^{137}Cs and their daughter isotopes.

The heat load of the waste is important for geologic disposal because it affects repository design. A geologic repository will likely have multiple temperature limits at different repository locations, and some rock types and repository designs are better at dissipating the heat than others. For example, waste packages disposed of in a geologic repository with host rock that has a low thermal conductivity would need to be placed farther apart than if the same waste packages were disposed of in a repository with host rock that has a high thermal conductivity. For a given total waste thermal output, reducing the volume of waste increases the thermal power density (e.g., watts/m^3) and vice versa. Therefore, the size of the repository footprint is, to a first approximation, driven by the total heat output of the waste and repository-specific temperature limits.

Another important consideration with respect to the thermal characteristics is the effect that increased temperature will have on degradation processes. In addition to impacting the repository footprint, increased thermal load has the potential to increase degradation rates for all components of the Engineered Barrier System. DOE has recently investigated the effect of higher thermal loads in multiple studies focused on the storage, transportation, and disposal of dual-purpose canisters.

3.4.1.3 Chemical Characteristics

Knowledge of the bulk and surface chemistry of spent fuels and waste forms in deep geologic disposal is also of paramount importance for repository performance. For any given disposal concept, bulk and surface chemistry dictate the reactivity and degradation rate of spent fuel and waste forms subjected to a specific set of environmental conditions. The lifetime of the waste forms in disposal can range from extremely long periods of time for reduced oxide waste (in repository concepts with reducing conditions without oxidant sources) to very short lifetime such as for salt waste (e.g., prospective waste from MSRs) exposed to humidity. For commercial LWR spent fuel, in the former case the waste form lifetime can be considered the limiting factor for the radionuclide source-term, while in the latter waste form degradation is considered instantaneous and waste package longevity becomes the limiting factor for radionuclide release.

Another important aspect of the chemical characteristics of the spent fuel/waste relates to interfacial processes. In particular, whether the degradation of spent fuel/waste can affect the bulk chemistry of solutions in the vicinity of the source term and eventually shorten the lifetimes of materials of the engineered barrier system (EBS) and other proximal waste forms. Some trace constituents from waste degradation might impact EBS or nearby waste forms/spent fuel. Strong exothermic reactions, for example, due to the pyrophoricity of Na-bonded metallic fuel, can negatively impact the local chemical environment in disposal, therefore rendering such waste unsuitable for direct disposal in geologic repositories. In contrast, interfacial chemistry near the waste form/spent fuel surface can also lead to the formation of secondary corrosion phases which may act as diffusion barriers and help delay waste

degradation and dissolution processes. Research on the prediction of secondary products at such interfaces is needed in the context of AR fuels.

A major concern is also to avoid any unfavorable chemical environment in the vicinity of the waste/spent fuel conducive to gas generation during the million-year post-closure period. In the context of high-level waste (HLW) waste/spent fuel, anoxic corrosion of steel canisters can produce large amounts of H_2 gas, which can result in gas-pressure buildup inside the sealed waste canisters and ultimate waste package failure detrimental to nuclear waste repositories. Processes such as water radiolysis due to ionizing radiation in the vicinity of spent AF/waste need to be assessed, since they can generate a significant amount of gas. Gas solubility data in complex multi-component salt waste envisioned to be produced by MSRs are needed for performance and safety assessment of salt waste in various repositories.

Finally, the waste treatment processes utilized at the BENFC prior to disposal should be compliant with the Resource Conservation and Recovery Act (RCRA). More specifically, particular attention should be paid to any chemical agents introduced as part of the waste treatment processes that could possibly be categorized as hazardous according to the RCRA and prohibited from land disposal under RCRA's Land Disposal Restrictions (40 CFR 268) unless a variance, extension, or exemption is obtained (EPA, 2001).

3.4.1.4 Physical Characteristics

The main physical characteristics of interest for handling and disposal of the waste/spent fuel are typically the physical state of the waste, its dimensions and mass, and the condition of the waste form/spent fuel. The physical state of the waste can be either solid, powder, or liquid. The dimensions and mass of the waste forms before packaging are important to assess their degradation rate and lifetime in disposal (dimensions and mass of the packaged waste forms are more relevant to handling during storage and transportation, and placement at the disposal site). The condition of the waste form/spent fuel (e.g., glass log, intact/breached cladding, fine-grained broken pieces, etc.) is also of interest to determine more accurately its lifetime once the package is breached, since the solvent-accessible surface area and overall morphology of the waste form/spent fuel can impact its degradation rate. Considerations regarding the microstructure (inner structure) of the waste forms/spent fuels can also be relevant to degradation rates and lifetime in disposal; the microstructure is strongly tied to waste form/fuel fabrication and processing or history (e.g., fuel burnup, radionuclide inventory, engineered layered microstructure in TRISO fuel, etc.). It is also worth noting that the swelling characteristics of some of the accident tolerant/advanced fuels (e.g., uranium silicide fuels, and fuels with Cr- and FeCrAl-coated cladding or SiC-based cladding) subjected to higher burnup and temperature are also desirable, since swelling might impact the microstructure and the radionuclide self-diffusion rates in spent AF/waste, which in turn might affect the rate of release of radionuclides to the environment after the waste package/canister is breached.

3.4.1.5 Packaging Characteristics

Depending on the type of repository considered, the dimensions of the waste package can directly impact the disposability of the waste forms/spent fuels. Previous studies have shown that the disposal options are strongly correlated with the packaging characteristics (Hardin *et al*., 2015; SNL, 2014a; SNL, 2014b). In the case of the deep borehole disposal concept, for example, the diameter of the waste package cannot exceed 40 cm (Freeze *et al*., 2016).

In addition to the dimensions, factors related to various waste package functions can affect waste form/spent fuel disposal. Indeed, the waste package may be the primary containment vessel for some waste forms (e.g., direct disposal of salt wastes) and might ultimately control the rate of release of radionuclides to the environment. In unsaturated and crystalline repositories, low waste package degradation rates are expected under both oxidizing and reducing conditions, while medium waste package degradation rate under reducing conditions typically occurs in clay/argillite (SNL, 2014a; SNL,

2014b). Very limited credit is anticipated for waste package degradation in salt repositories, and in deep boreholes waste package containment might need to be maintained for 10 years under downhole conditions.

The waste package might also include provisions to reduce the potential for post-closure criticality including SNF loading criteria, geometry controls, neutron absorbers (e.g., in the form of plates), moderator control (e.g., fillers). Finally, the waste package design might provide an additional flexibility for the final disposal of some wastes/spent fuels that present more challenges for disposal site selection and repository design.

3.4.1.6 Safeguards and Security

Similar to safeguards for existing LWR spent nuclear fuel, radionuclide-containing waste generated from advanced nuclear fuels will be subject to safeguards through the operating life of the deep geologic repository. Because AR Fuels and advanced reactor waste streams (ARWS) may have differing characteristics relative to LWR fuel, it is worth evaluating if there will be an effect on the level of safeguards after repository closure. While safeguards and security for AR SNF will be precisely defined only after AR fuel cycles have been fully established (see, e.g., Riley *et al*., 2018), it can be inferred that newly developed safeguards will likely share the central concept of "self-protection" already in application for LWR spent fuel. According to current NRC regulations in 10 CFR Part 73, self-protection is attributed to SNF "…which is not readily separable from other radioactive material, and which has a total external radiation dose rate in excess of 100 rem per hour at a distance of 3 ft from any accessible surface without intervening shielding." With the present regulations, if commercial LWR spent fuel is no longer self-protecting, such as for spent fuel assemblies containing more than 2 kg of plutonium (i.e., Category I safeguard and security), enhanced security and safeguards are required. For AR SNF, the minimum handling unit that might be applied for a future self-protection criteria may differ from that for LWR, or may not as of yet be determined.

In addition, several other repository characteristics shared with LWR spent nuclear fuel could also impact security measures for AR SNF and waste forms, such as the ease of adversary access to mined repositories or deep borehole disposal sites, or the length of time for intrusions by adversaries in the different generic repositories. From the viewpoint of security, access and operation/maintenance of mined repositories in either soft or hard media or of deep boreholes are very different. For example, whereas clay/argillite or salt repositories feature vertical shaft access, access to crystalline repositories is generally provided by vehicles, whereas extremely limited access is available for deep borehole disposal. Finally, due to the limited access to perform any maintenance in deep borehole disposal, it is likely that boreholes will be sealed soon after waste placement, in contrast with disposal in mined repositories, which allows for longer operation and maintenance. These different characteristics will need to be accounted for when defining security measures for disposal of AR SNF and waste forms in the various generic repositories.

3.4.1.7 Summary of Desirable Waste Characteristics by Repository Concept

Some of the desirable generic waste characteristics in terms of radionuclide inventory, thermal output, chemical reactivity, physical aspect and dimensions, packaging, and safeguards and security are discussed above in Sections 3.4.1.1 through 3.4.1.6 are summarized in Table 3-8 for the five main generic disposal concepts, namely, unsaturated, crystalline, clay/argillite, salt and deep borehole disposal.

Table 3-8 Summary of Desirable Characteristics of Waste by Generic Repository Concept

	Unsaturated	Crystalline	Clay/Argillite	Salt	Deep Borehole
Inventory	Quantity and concentration of long-lived fission products and activation products. (^{14}C, ^{36}Cl, ^{79}Se, ^{93}Zr, ^{99}Tc, ^{107}Pd, ^{126}Sn, ^{129}I, ^{135}Cs)				
	Fissile quantity & concentration for criticality calculations	Fissile quantity & concentration for criticality calculations	Fissile quantity & concentration for criticality calculations	Criticality calculations might be limited to highly-enriched fuels	Fissile quantity & concentration for criticality calculations
	Safeguards Concerns Addressed				
	Redox- sensitive & anionic radionuclides relatively mobile	Anionic radionuclides relatively mobile	Anionic radionuclides relatively mobile	Low mobility for all radionuclides	Low mobility for all radionuclides
Thermal Characteristics	High thermal loading	Low thermal loading	Low/medium thermal loading	High thermal loading	High thermal loading
	Information on potential phase transitions induced by radioactive decay heat				
Chemical Characteristics	Not pyrophoric or otherwise vigorously reactive to water				
	Moderate waste form degradation rate	Low waste form degradation rate	Medium waste form degradation rate	High waste form degradation rate	High waste form degradation rate
	Not gas generator	Not gas generator	Not gas generator		Not gas generator
	Meets Resource Conservation and Recovery Act requirements				
Physical Characteristics	Moderate waste form degradation rate	Low waste form degradation rate	Medium waste form degradation rate	Medium to high waste form degradation rate	High waste form degradation rate
	Any size packages	Only small packages	Only small packages	Up to medium-size packages	Only very small packages
Packaging	Low waste package degradation rate under oxidizing conditions	Low waste package degradation rate under reducing conditions	Medium waste package degradation rate under reducing conditions	Very limited credit for waste package degradation	Waste package containment maintained for 10 years under downhole conditions
					Minimize interior void space
	Any size packages	Only small packages	Only small packages	Up to medium-size packages	Only very small packages
	Low probability of post-closure criticality	Low probability of post-closure criticality	Low probability of post-closure criticality	Highly unlikely post-closure criticality for LEU Low probability of post-closure criticality for HALEU	Low probability of post-closure criticality
	Delivered to repository in containers designed to be re-opened				
Safeguards & Security	Consider concept of self-protection and access of adversary to waste				

3.4.2 Preliminary Assessment of Potential Disposal Pathways for Advanced Reactor Fuels and Advanced Reactor Waste Streams

A preliminary assessment of gaps for the disposal of spent AFs and associated waste reveals that, in comparison to standard LWRs, which have known types of waste, disposition strategies for most spent AFs and waste produced from the utilization of AFs are not currently as well-defined. This stems in part from the fact that most ARs are not expected to be commercially deployed until *ca*. 2030.

Other factors, such as the protection of proprietary information on innovative AFs under development by various advanced nuclear reactor and fuel companies, also contribute to the lack of clear disposal strategies. Proprietary information for AR design may include important details about the thermal, physical, and chemical characteristics of AFs. Lacking such details, analysis of tenable disposal pathways is incomplete, tentative, or impossible.

Additionally, there is also currently considerable uncertainty regarding the operating conditions, refueling needs, and performance of some of AR concepts, which can significantly affect the type and quantity of waste generated. For example, one can consider the case of MSRs, whose current level of technological readiness is at the level of engineering demonstration/proof of concept with the Aircraft Reactor Experiment and MSRE in the U.S. MSRs can be configured and/or operated in many ways (i.e., they can operate as breeders, use epithermal, thermal, or fast neutron spectra, utilize either solid or liquid fuels, and use a variety of fuels mixtures including thorium and ^{233}U, highly-enriched uranium (HEU), ^{239}Pu, low-enriched uranium (LEU), or transuranic (TRU) elements from spent fuels).

As a result of this preliminary assessment for the disposal of spent AFs and their associated waste, the primary gap is that the waste streams/waste forms are relatively poorly defined. Again, this is owing to the lack of commercial deployment. For TRISO and metallic fuels, precedent waste forms with similar characteristics have at minimum been analyzed for direct disposal scenarios. Sodium-bonded metallic fuels and MSR-derived fuels and wastes streams, there is not such direct disposal precedent pathways (see Section 5 for further analysis of Disposal Pathways).

Consideration of processed waste streams adds complexity to feasibility analysis and gap analysis. This is because uncertainty is introduced as the complexity of post-reactor operations and handling of spent fuel and it associated products incorporates additional steps and considerations beyond those for direct disposal. This is not to say that there is lack of technical feasibility, but rather there is insufficient information or definition of all relevant parameters to fully analyze all aspects of a given fuel disposal pathway.

The wastes produced by fuel fabrication and fuel utilization for any AR fuel cycle should also be assessed directly in comparison to the current LWR fuel cycle, for wastes that are the same/closely similar materially/chemically. This is the case for LWR small modular reactors, whose waste management will be similar to current LWR systems. In addition, information regarding the material/chemical/physical form of the intended waste form for each waste stream from each AR fuel cycle is essential R&D, including information regarding dimensions, mass, and containerization intended. Direct interaction with reactor designers is important in waste stream characterization and waste form design. For example, as mentioned above, dimensions of the waste/spent fuel can be important if the deep borehole disposal concept is considered. In addition, a fertile arena for R&D exists in constraints on waste forms that are being newly proposed in terms of characterization, as well as constraining degradation behavior and waste form lifetime.

In order to realistically assess disposal of AR SNF / AR waste and identify waste characteristics and metrics, a down selection of waste streams and waste forms to potential representative AR fuel cycles is

necessary. Under DOE's Generation IV (GenIV) program in the 1990s, six classes of reactor technologies were selected, namely the MSR, very high temperature gas–cooled reactor (VHTR), sodium-cooled fast reactor (SFR), lead-cooled fast reactor, gas-cooled fast reactor, and supercritical LWR. The choice of these AR concepts during GenIV was based on a set of requirements to meet future needs for safe, sustainable, environmentally responsible, economical, proliferation-resistant, and physically secure energy (INL, 2005). Among these reactor concepts, most support by DOE has gone initially to the development of SFR and VHTR, and more recently to MSR.

For this reason and for the sake of simplicity, in this study only waste streams and waste forms are considered for the following fuel cycles:

- TRISO-type fuel which could originate from multiple reactor sources - VHTR, gas-cooled fast reactor, fluoride salt-cooled MSR, or some microreactors using LEU or HALEU
  - once-through, direct disposal
- Sodium Fast Reactor (SFR) - two cases considered
  - pyroprocessing recycle
  - once-through, treated
- Salt-Fueled MSR
  - once-through, treated

The waste stream and waste form classification in are analyzed for these four cases. The different levels of technological readiness (i.e., "well known", "conceptually known", "unknown"), and whether an item is included in the suite of tables for disposal considerations or an item is part of optional process are shown in Table 3-9 through Table 3-12. While low-level waste (LLW) in the following tables is not being considered here for assessment for the various deep geologic repository concepts, GTCC LLW is included in this study by analogy with the current management of some of those materials as HLW for disposal.

The TRISO-type waste streams and possible related waste form classification, are shown in Table 3-9.

Table 3-9 Waste stream and waste form classification for TRISO-type AR fuel cycle.

TRISO-Type Fuel Cycle (LEU, once-through, direct disposal)	
Waste Stream	**WF Classification**
Process—Kernel fabrication and Kernel TRISO	
From kernel fabrication: water(sludge), hot gas, reject kernels/kernel fabrication *From kernel TRISO coating and pressing,* hot gas, reject particles, and pressed fuel	LLW (Note that rejected kernels, reject particles, and pressed fuel could be considered special nuclear materials)
Process—Reactor	
Graphite superstructure	LLW
Spent Fuel	WF 1: SNF

LEGEND:

Color coding for knowledge base	Well known	Conceptually known	Unknown

List of Acronyms: GTCC: Greater Than Class C; HLW: High-Level Waste; LLW: Low-Level Waste; L-L FP: Long-Lived Fission Products; MS: Molten Salt; S-L FP: Short-Lived Fission Products; TRU: TRansUranic; WF: Waste Form.

The waste stream associated with kernel fabrication produces waste such as water (sludge), hot gas, reject kernels/kernel fabrication, while kernel TRISO coating and pressing leads to hot gas, reject particles, and pressed fuel, with all the waste produced expected to be LLW. Waste generated at the backend of the TRISO-type fuel cycle consists essentially of: (1) spent TRISO fuel particles embedded in the inner region of graphite spherical pebbles for AR designs using TRISO-bearing fuel pebbles; or (2) graphite Class C LLW from prismatic-block-type fuel, which typically consists of a solid graphite block containing both coolant channels and fuel channels, and graphite compacts in which spent TRISO fuel particles are embedded for ARs using TRISO-bearing prismatic block fuel.

For the second aspect above, it is assumed that the fuel compacts filling the fuel channels can be readily extracted from the graphite prismatic block to reduce the overall volume of the waste to be disposed of in repositories or deep boreholes (Lotts et al., 1992). The main processes envisioned for the separation of the fuel compacts from the graphite block are either physical/mechanical separation (e.g., pushing the fuel compacts out of the channels or crushing the graphite block) or chemical process (e.g., burning the graphite matrix, with concomitant production of CO_2) (Lotts et al., 1992).

Since the TRISO particles are bonded together into graphite compacts, there is usually no option/need to recover individual spent TRISO particles before disposal. Direct disposal of whole graphite prismatic blocks and pebbles containing spent TRISO-fuel is often considered the preferred option, owing to its lower cost, ease of implementation, and limited safeguards requirements and proliferation risk (Lotts *et al*, 1992; IAEA, 2013; Hall, 2019a).

The Sodium Fast Reactor (SFR; pyroprocessing recycle) waste streams and possible related waste form classification, are shown in Table 3-10.

Table 3-10. Waste stream and waste form classification for Sodium-Cooled Fast Reactor Metal Fuel Cycle (pyroprocessing recycle) AR fuel cycle.

Sodium-Cooled Fast Reactor Metal Fuel Cycle (pyroprocessing recycle)	
Waste Stream	**WF Classification**
Process—Fuel fabrication	
• Cladding • Bond sodium • Assembly hardware (Assume fuel scrap recycled)	WF 1: GTCC LLW (L-L TRU) similar to defense TRU (Does Not Exist Currently)
Process—Pyroprocessing/ electrochemical processing	
Compressed gas bottles	WF 2: Noble gases released after sufficient decay
Fuel hulls and noble metal fission products	WF 3: HLW (S-L FPs)
• Electrorefining salt • Bond sodium	WF 4: HLW (all)
Off-gas from electrorefining salt	LLW
Process—Reactor	
End-of-life waste stream	WF 5: SNF (Note Na- bonded SNF Not Directly Disposable)

LEGEND:

Color coding for knowledge base	Well known	Conceptually known	Unknown

List of Acronyms: GTCC: Greater Than Class C; HLW: High-Level Waste; LLW: Low-Level Waste; L-L FP: Long-Lived Fission Products; MS: Molten Salt; S-L FP: Short-Lived Fission Products; TRU: TRansUranic; WF: Waste Form.

The waste stream associated with fuel fabrication produces waste from the cladding, bond sodium or assembly hardware, with the corresponding waste form being GTCC LLW (WF 1. GTCC LLW (L-L TRU)) similar to defense TRU. For the pyroprocessing/electrochemical processing, four main waste streams are expected, which are associated with: (1) compressed gas bottles, (2) fuel hulls and noble metal fission products, (3) electrorefining salt and bond sodium, and (4) off-gas from electrorefining salt. The corresponding waste forms are: Noble gases released after sufficient decay (WF 2), HLW (WF 3. HLW (S-L FPs) and WF 4. HLW (all)), and LLW. The end-of-life waste stream is associated with SNF (WF 5. SNF) that is not directly disposable due to the pyrophoric nature of sodium-bonded SNF.

The SFR (once-through, treated) waste streams and possible related waste form classification, are shown in Table 3-11.

Table 3-11. Waste stream and waste form classification for Sodium-Cooled Fast Reactor Metal Fuel Cycle (once-through, treated) AR fuel cycle.

Sodium-Cooled Fast Reactor Metal Fuel Cycle (once-through, treated)	
Waste Stream	**WF Classification**
Process—Pyroprocessing-electroreduction of oxide feedstock to produce metal feedstock (optional)	
Compressed gas bottles	Noble gases released after sufficient decay
Fuel hulls and noble metal fission products	HLW (S-L FPs)
Electrorefining salt	HLW (all)
Off-gas from electrorefining salt treatment	LLW
Process—Fuel fabrication	
• Cladding • Bond sodium • Assembly hardware (Assume fuel scrap recycled)	WF 1: GTCC LLW (L-L TRU) similar to defense TRU *(Does not exist currently)*
Process—Reactor	
• Used driver fuel • Used blanket fuel • Used homogeneous core fuel	WF 2: SNF (*Note Na-bonded SNF Not Directly Disposable*)
Process—Treatment (melt-dilute) (treatment may vary depending on disposal concept)	
Treated SNF in metal	WF 3: HLW (2 WF)
Filters from off-gas from melt-dilute treatment	LLW

LEGEND:

Color coding for knowledge base	Well known	Conceptually known	Unknown

List of Acronyms: GTCC: Greater Than Class C; HLW: High-Level Waste; LLW: Low-Level Waste; L-L FP: Long-Lived Fission Products; MS: Molten Salt; S-L FP: Short-Lived Fission Products; TRU: TRansUranic; WF: Waste Form.

For Once-through Sodium Bonded Metallic Fuel. the first process – which may be optional – is the pyroprocessing/electroreduction of oxide feedstock to produce metal feedstock. This process is characterized by waste streams and waste forms identical to the pyroprocessing/ electrochemical processing of Table 3-10 (see WFs 2 through 4 and LLW). The fuel fabrication processes in Table 3-11 and Table 3-10 are also identical, with similar waste streams and WFs (see Table 3-11 WF 1). Similarly, WF 2 (the same as Table 3-10 WF 5) the end-of-life waste stream (i.e., used fuel) of sodium-bonded fuel would not be directly disposed and would need to be treated via pyroprocessing

(electrometallurgical treatment, EMT), or the melt dilute process. Those processes would lead to the same two WF remaining for this SCFRMFC, Table 3-11 WF 3, which includes both the metallic waste form (same as Table 3-10 WF 3) and the CWF (same as Table 3-10 WF 4 and planned to be made from the EMT salt waste).

The MSR waste streams and possible related waste form classification, are shown in Table 3-12.

Table 3-12. Waste stream and waste form (WF) classification for Salt-fueled Molten Salt Reactor (once-through, treated) AR fuel cycle.

Salt-Fueled Molten Salt Reactor Fuel Cycle (once-through, treated)	
Waste Stream	**WF Classification**
Process—Fuel fabrication	
Filters from treating off-gas	LLW
Process—Reactor	
Filters for insoluble fission products	WF 1. GTCC LLW (S-L FPs)
Caustic scrubber effluent	WF 2. GTCC LLW (all)
Silver functionalized absorbent	WF 3. GTCC LLW (L-L FPs)
Molecular sieve	LLW
Metal organic framework absorbent	WF 4. Gas released after decay storage
Process—Treat MS for disposal (three possible options)	
Mixture of metal chloride or fluoride salts	WF 5a. OPTION 1 HLW (all) (phosphate glass)
Mixture of metal chloride or fluoride salts	WF 5b. OPTION 2 HLW (all) (ceramic-metal composite)
Mixture of metal chloride or fluoride salts	WF 5c. OPTION 3 HLW (all) (glass-bonded ceramic)
Off-gas	LLW

LEGEND:

Color coding for knowledge base	Well known	Conceptually known	Unknown

List of Acronyms: GTCC: Greater Than Class C; HLW: High-Level Waste; LLW: Low-Level Waste; L-L FP: Long-Lived Fission Products; MS: Molten Salt; S-L FP: Short-Lived Fission Products; TRU: TRansUranic; WF: Waste Form.

As shown above, aside from LLW generated from filters from treating off-gas after fuel fabrication, four main waste streams and associated waste forms have been identified for the salt-fueled MSR fuel cycle once-through: Filters for insoluble fission products (WF 1. GTCC LLW (Short-Lived Fission Products (S-L FPs))); Caustic scrubber effluent (WF 2. GTCC LLW (all)); Silver functionalized

absorbent (WF 3. GTCC LLW (Long-Lived Fission Products (L-L FPs))); Metal organic framework absorbent (WF 4. Radioactive gas released after decay storage).

If additional treatment of molten salt spent fuel is utilized for disposal purposes, a fifth waste form can be envisioned from the waste stream consisting of mixtures of metal chloride or fluoride salts containing unknown amounts of radionuclides (see Table 3-12, WF Classification column, 5a, 5b, and 5c)).

Depending on the options/concepts used for treating molten salt spent fuel, three main types of HLW WF may result from treatment: phosphate glass (WF 5a. Option 1); Ceramic-metal composite (WF 5b. Option 2); glass-bonded ceramic (WF 5c. Option 3). In comparison to WF 1 through 4 discussed above, it can be inferred that WF 5a-c will have relatively longer lifetimes in the generic disposal environments under consideration, although knowledge gaps remain and WF performance should be further investigated. For example, more constraints are needed to assess the degradation of metallic components of WF 5b under oxidizing unsaturated conditions and understand the impact of local redox conditions on WF degradation in the various mined repository concepts. The lack of information on the radionuclides inventory and fissile loading from the MSR spent nuclear fuel significantly complicates the assessment of WF 5a-c for the disposal options under consideration, in particular medium to high issues related to WF thermal output and criticality and their impact on the various disposal concepts.

4 BACK END OF THE NUCLEAR FUEL CYCLE CONSIDERATIONS

This section begins with an overview of existing storage and transportation experience and then identifies the storage and transportation challenges of the back end of the nuclear fuel cycle (BENFC) associated with accident tolerant fuels (ATFs) and advanced fuels (AFs). The section ends with discussion of disposal challenges.

4.1 Existing storage experience

The U.S. currently utilizes a once-through, open fuel cycle that begins with on-site interim storage of spent nuclear fuel by the industry utilities and is planned to end with permanent disposal of the spent fuel in a deep geologic repository. The open fuel cycle in the U.S. is currently in process, but lacks an existing/operating repository. The historical context and timeline leading to the U.S. decision to use an open fuel cycle is comprehensively covered in the NASEM (2022) report, "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors".

Freeze et al. (2021), in the report "Integration of the Back End of the Nuclear Fuel Cycle," provide a detailed overview of the current practices for the storage and transportation of existing waste streams. In the report, it was noted that spent nuclear fuel (SNF) from commercial reactors was initially planned to be stored in spent fuel pools (SFPs) and moved directly into permanent disposal. However, by the early 1990s, as SFPs began to reach their capacity limits and delays in disposal implementation, the nuclear reactor utilities moved to dry storage at their facilities. This section will summarize wet versus dry storage and the components of storage canisters used to store the current fleet of spent fuel.

4.1.1 Wet storage

Nuclear fuel is considered "spent" after about four to six years of reactor operation. Although the fuel is no longer able to efficiently produce energy, it is still thermally hot, and it emits radiation. To protect workers at reactor sites from this radiation, newly discharged fuel assemblies are transferred to SFPs in what is known as wet storage of spent fuel (Figure 4-1).

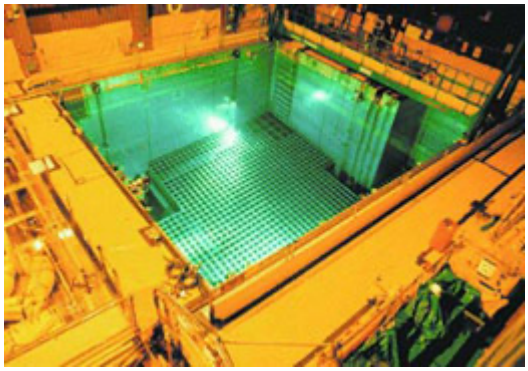

Figure 4-1 Spent fuel pool for storage of newly discharged spent fuel (NRC, 2020b).

The assemblies are stored in metal racks within the SFPs. Typically, SFPs are at least 12 meters (40 feet) deep, allowing the assemblies to be covered by at least 6 meters (20 feet) of water, which provides radiation shielding (Freeze et al., 2021; NRC, 2014a). Spent fuel assemblies are moved into SFPs from the reactor along the bottom of water canals, maintaining radiation shielding throughout the process of assembly transfer to the SFP, where they reside for at least three years, but typically for at least 10 years

(Freeze et al., 2021; Hanson & Alsaed, 2019). Higher burnup spent fuel, due to the extraction of more power from the fuel in a reactor before fuel replacement, has higher fission product content and entails a few more years of cooling (Freeze et al., 2021; NRC, 2014a; NASEM, 2022). The SFPs are also used when transferring/loading spent fuel into dry storage canisters.

The general strategy for a commercial reactor is to utilize the entire wet storage capacity while accounting for/reserving the space for a full reactor core discharge (Freeze et al., 2021; Rechard et al., 2015). Once the wet storage at a reactor site is fully utilized, spent fuel in the pool must be moved into dry storage to accommodate newly discharged fuel.

A typical SFP can hold anywhere from about four to ten reactor core loads, or about 400 to 1200 metric tons of heavy metal (MTHM), and these maximum loadings were established as a result of NRC approval of packing the spent fuel assemblies more densely in the pools to increase SFP capacity (Freeze et al., 2021; StoreFUEL, 2021). The amount of time for an SFP to reach full capacity is 35 years for low-burnup operation and 46 years for high-burnup operation (Freeze et al., 2021; NRC, 2014a).

At the end of 2016, the spent fuel pool inventory was 50,195 MTHM (see Table 2-2 in Freeze et al., 2021), but by 2019 there were an estimated 44,391 MTHM of spent fuel corresponding to 155,342 fuel assemblies in wet pool storage at operating and shutdown reactor sites (see Table 2-1 in Freeze et al., 2021). This indicates a decrease in assemblies stored in SFPs as reactors shut down and spent fuel continues to be transferred to dry storage (see Figure 2-3 in Freeze et al., 2021).

4.1.2 Dry storage

Dry storage of SNF follows from wet storage, and was initiated in the U.S. following the realization in the mid-1980s that spent fuel pools would have space-limitation issues for storage of spent fuel (NASEM, 2022).. The U.S. uses multiple dry cask storage systems (DCSSs), with the most common systems being canister-based. The canister-based systems involve sealing multiple fuel assemblies in a large, stainless-steel canister (e.g., a dual-purpose canister (DPC)). Once the spent fuel in SFPs needs to be transferred to dry storage, the assemblies are loaded into baskets inside dry storage canisters within the SFPs. After assembly loading, the canisters are removed from the SFPs and are dried using a commercial drying procedure. This procedure begins with draining enough water from the canisters (using a drain tube) to seal the canisters using welded lids. The canisters are then decontaminated. From there, vacuum drying or forced helium dehydration (using pressurized helium to force water out of the canister) is applied where the decay heat of the assemblies is utilized to remove residual water from the cask. Once the residual water has been sufficiently removed according to the drying adequacy criteria in NUREG-2215 (NRC, 2020a), the canisters are backfilled with helium. These canisters are then placed in a concrete or metal storage cask at the interim storage location (Figure 4-2) – the storage cask is either welded or bolted shut (Hanson & Alsaed, 2019; Freeze et al., 2021).

Non-canistered systems (e.g., bare fuel casks) are alternative options for dry storage – these systems involve loading spent fuel assemblies into baskets that are directly integrated into a metal cask. When loading bare fuel casks, the casks themselves are placed into the SFP for loading. After the assemblies are loaded, the cask is sealed and removed from the pool. Water is drained using a drain tube, and the cask is then transferred to the drying location, where either vacuum or forced helium dehydration is applied (Hanson & Alsaed, 2019; Freeze et al., 2021).

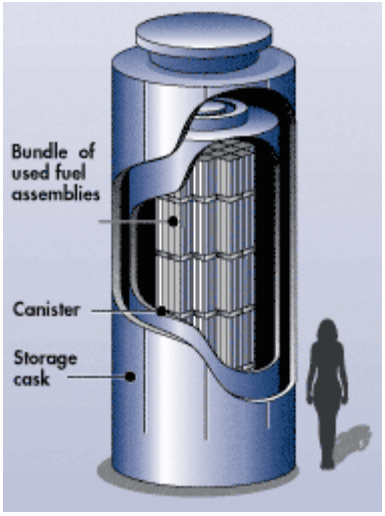

Figure 4-2 Spent fuel in a canister-based dry storage system (NRC, 2020c).

DCSSs can have either vertical or horizontal configurations. Vertical DCSSs can be aboveground or belowground, with the majority being aboveground (Hanson & Alsaed, 2019). Horizontal DCSSs are stored in concrete "vaults" (Figure 4-3).

(a) Vertical Above Ground
(Maine Yankee, Wiscasset, ME)

(b) Vertical Below Ground
(Humboldt Bay, Eureka, CA)

(c) Horizontal
(Rancho Seco, Herald, CA)

Figure 4-3 Dry Cask Storage Systems (Freeze et al., 2021; Rechard et al., 2015).

Within the DCSSs, passive cooling occurs via natural convection in the annular region between the canister and the overpack in vertical systems and between the canister and vault in horizontal systems. This reliance on passive cooling makes DCSSs less vulnerable to system failures. The largest DCSSs hold 37 PWR assemblies and 89 BWR assemblies (Freeze et al., 2021).

By the end of 2019, there was an estimated 39,207 MTHM (134,307 assemblies) of commercial spent fuel in 3,158 DCSSs at operating and shutdown reactor sites. The 3,158 DCSSs include DPCs and bare fuel casks with commercial spent fuel. There are also 29 casks with DOE-managed spent fuel and 16 casks with Greater Than Class C waste, for a total of 3,203 DCSSs (Freeze et al., 2021).

DCSSs are located on Independent Spent Fuel Storage Installations (ISFSIs). These ISFSIs are most commonly co-located with reactors. 10 CFR Part 72, which covers NRC general requirements for dry storage of spent fuel, authorizes construction and operation of ISFSIs by general and site-specific licenses. General licenses (10 CFR Part 72, Subpart K) can be used by a nuclear power plant licensee to

store spent fuel in NRC-approved casks at a site license to operate a reactor under 10 CFR Part 50. Site-specific licenses (10 CFR Part 72, Subparts A through I) can be used at or away from a reactor site based on the ISFSI application, which contains technical requirements and operating conditions (Freeze et al., 2021).

As of November 2020, there were 81 ISFSI licenses, with 66 of those being general licenses and the other 15 being site-specific licenses. Of the 66 general licenses, 51 are at operating reactor sites and 15 are at shutdown reactor sites with stranded waste. Of the 15 general licenses, seven are at operating reactor sites, three are at shutdown reactor sites with stranded waste, and five are at away-from-reactor locations (Freeze et al., 2021).

A license application SAR for an ISFSI, according to 10 CFR 72.24(d), requires "…an analysis and evaluation of the design and performance of structures, systems, and components (SSCs) important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI." (Hanson & Alsaed, 2019). NUREG-2215 (NRC, 2020a) groups the SSCs important to safety into the following categories:

- Confinement canister (shell and associated welds and bolts)
  - Fuel basket
  - Fuel and cladding
  - Racks for positioning stored fuel or waste material within the canister or cask (including lifting components)
  - Closure lids
  - Closure welds
- Transfer cask
- Storage overpack (horizontal, vertical, or underground)
- Storage cask

The general safety functions of the ISFSI, according to 10 CFR Part 72, are "(1) To maintain the conditions required to store spent fuel…safely; (2) To prevent damage to the spent fuel… [and] waste container during handling and storage; or (3) To provide reasonable assurance that spent fuel… can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public." (Hanson & Alsaed, 2019). NUREG-2215 (NRC, 2020a) states "[The] SCCs may provide confinement, subcriticality, radiation shielding, support, and retrievability safety functions of the stored materials, and therefore, should be appropriately maintained under all credible loads and their combinations for normal, off-normal, and accident conditions and natural phenomena effects." These SSCs and their safety metrics can serve as a foundation for the storage considerations of existing waste streams.

4.2 Existing transportation experience

Spent fuel in SFPs and DCSSs, at some point in time, will need to be transported to either an interim storage facility or a geologic disposal site (Hanson & Alsaed, 2019). In the U.S., truck or rail options have been the typical transportation pathways; other nations ship spent fuel (Freeze et al., 2021; BRC, 2012). There is plenty of transportation experience for existing waste streams, with transportation having occurred on an intermittent basis since the early 1960s. Transportation campaigns since this period have involved the return of foreign spent fuel to the U.S., shipping of experimental spent fuel for testing, spent fuel transfers between reactors owned by the same company, and shipping of naval spent fuel for storage).

More than 3,000 shipments have been made in the U.S. for a total of ~1.7 million miles of transport (Freeze et al., 2021; Rechard et al., 2015; DOE, 2009).

Transportation of spent fuel involves removal of the canister containing spent fuel assemblies from a storage cask using a transfer cask and subsequent emplacement in a reusable, shielded transportation cask. Like storage casks, transportation casks provide structural protection and containment, radiation shielding, and thermal management. About 85% of commercial spent fuel in dry storage has certified transportation casks readily available, but the existence of storage-only canisters complicates future transportation campaigns (Freeze et al., 2021).

Transportation is likely to be by rail due to the large diameter and weight of the rail casks, which have a typical diameter of 2.4 meters, an overall diameter (including impact limiters) of 3.4 meters, and an overall length (including impact limiters) of 7.6 meters; rail cask systems with loaded DPCs may weigh anywhere between 113.4 and 158.8 metric tons (Freeze et al., 2021; Greene et al., 2013). The largest rail casks can accommodate canisters with up to 37 PWR assemblies or 89 BWR assemblies (Freeze et al., 2021), corresponding to the largest canisters in DCSSs. Meanwhile, truck casks can only accommodate smaller canisters (up to four PWR or nine BWR assemblies) and have a loaded weight of about 23 metric tons (Freeze et al., 2021; Greene et al., 2013). Their primary utilization would be for older shutdown reactor sites that lack rail access (Freeze et al., 2021).

NRC regulations for spent fuel transportation can be found in 10 CFR Part 71. Transportation packages must, under normal conditions of transport and hypothetical accident conditions (HAC), have “no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging.” In addition, “no accessible surface of a package would have a temperature exceeding 50 °C” (Freeze et al., 2021; 10 CFR Part 71.43).

NRC certifies transportation cask designs through numerical modeling results of HAC, which have been validated with scaled testing (Freeze et al., 2021; Rechard et al., 2015). According to the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (10 CFR Part 961), nuclear utilities are responsible for preparing commercial spent fuel for shipment in transportation casks provided by DOE (Freeze et al., 2021).

NRC performed a risk assessment for spent fuel transportation that examined doses and risks from normal conditions of transport and from two HAC scenarios (NRC, 2014b). The results of this assessment affirmed that radiological consequences from spent fuel transportation conducted in compliance with NRC regulations are low, and that these regulations are sufficient to protect the public against unreasonable risk (Freeze et al., 2021; NRC, 2014b). Overall, the regulations that govern the transport of spent fuel, HLW, and other nuclear materials have resulted in an excellent safety record (Freeze et al., 2021; BRC, 2012).

4.3 Storage and Transportation Challenges

The storage and transportation challenges for the current fleet of SNF were identified and prioritized according to both the timing of the need for information and the importance of this information to licensing in PNNL-28711, “Gap Analysis to Support Extended Storage and Transportation of Spent Nuclear Fuel: Five-Year Delta.” (Hanson & Alsaed, 2019). A separate high-level analysis was conducted to address the storage and transportation challenges associated with ATFs and AFs. (Honnold et al., 2021). The challenges listed in this section are derived from collating these two analyses, which incorporate both prior experience and new knowledge focused on ATF/AF and maintain the priority levels identified in the high-level ATF/AF gap analysis report.

4.3.1 High-Priority Storage and Transportation Challenges

4.3.1.1 Thermal Profiles

Nearly all degradation mechanisms for storage and transportation materials and structures are temperature dependent. NUREG-2215 (NRC, 2020a) defines a 400 ℃ cladding temperature limit for storage. For transportation, 10 CFR 71.43(g) states that there is a 50 ℃ thermal limit on accessible surfaces for transportation packages in a nonexclusive use shipment and an 85 ℃ thermal limit for these types of packages in an exclusive use shipment.

Although the ATF/AF designs may have improved in-reactor performance at higher temperatures, the effect of potential higher temperatures (higher burn-up/more fission products) during storage and transportation may benefit from being validated with experimental and analytical data. For ATF/AF storage, the 400 ℃ cladding temperature limit may not be appropriate to modified or new systems. Significantly different fuel forms (e.g., TRISO) may be characterized with different tests to determine longer-term mechanical integrity of this spent fuel under various thermal regimes (including higher temperatures). For example, thermal decomposition of the SiC layer in TRISO-based fuels has been identified as a potential failure mechanism (Hall *et al*., 2019c).

4.3.1.2 Stress Profiles

Stresses associated with normal transportation cask handling as well as design basis accidents including cask drop, seismic events, and cask tip-over could potentially cause cladding failure if the cladding has been sufficiently thinned due to creep.

ATF/AF concepts are expected to have different load capacities and ductile/brittle behaviors compared to the current fleet during handling operations, extended storage, and transport. ATF/AF cladding technologies are designed to reduce oxidation and thus reduce production and uptake of hydrogen, so stress profile-related concerns such as cladding oxidation, embrittlement, and reorientation of hydrides as well as delayed hydride cracking are assumed to be low importance. However, while the applied transportation loads are low enough for ATF/AF to not be of significant concern, the condition and geometry of the material may change from these stresses. Therefore, the effects of stress profiles must be evaluated for each individual concept.

4.3.1.3 Drying Issues

Many degradation mechanisms in storage and transportation systems are dependent on or accelerated by the presence of water. During the transfer of spent fuel from pool storage to dry storage, the spent fuel is placed in a dry storage canister and a vacuum drying procedure is conducted on the canister. Jung et al., (2013) investigated criteria for adequate dryness. The report states that a canister containing below 1 L (55.5 moles) of water would not see significant degradation of fuel, cladding or other internal components over 300 years. Water in excess of 1 L could result in a number of degradation mechanisms not currently considered. In addition, approximately 20 L of residual water may reach the flammability limit of 4% H_2 within 16 years, whereas 3-4 L of water could reach that limit within 300 years. NUREG-2215 (NRC, 2020a) references PNL-6365 (Knoll & Gilbert, 1987) to define a drying adequacy criterion that is met if a canister can maintain a vacuum hold less than or equal to 3 torr for 30 minutes. If the criterion is met, the canister is expected to have 0.43 moles of water remaining in the canister following the vacuum drying procedure, which is significantly lower than the thresholds set by Jung *et al*., (2013).

While some ATF/AF cladding concepts are intended to limit oxidation by water, the impact of residual water in dry storage on the waste forms of interest (and particularly in failed SNF) should be quantified.

The effectiveness of drying procedures on water removal is expected to be different across ATF and AR SNF concepts.

4.3.1.4 Fuel Transfer Options

The phenomena associated with wet and/or dry fuel retrieval options has been studied in detail for the current SNF fleet, where the primary concern is the behavior of hydrides in the cladding when exposed to changing environments during the transfer process. For example, if a cask were to be reopened in a pool, rewetting and rapid cooling of the cladding may quench hydrides in place, so their size and distribution would be different than those seen in dry storage. In addition, fuel assemblies would have to be re-dried to be returned to storage or for transportation, and multiple drying cycles could exacerbate hydride reorientation or creep and affect cladding performance. However, the effects of re-wetting/re-drying on hydrides is inconclusive due to the high sensitivity of this degradation mechanism to hoop stress.

ATF, especially those with higher densities, will have different physical characteristics (e.g., increased mass) that need to be assessed relative to plant and transportation infrastructure and operational constraints. As an example, heavier loadings (due to higher density fuels) during fuel transfer operations may exceed facility crane capabilities. Another example would be how brittle cladding types may not be capable of sustaining current handling loads without gross ruptures. Higher burnup fuel will have a higher temperature and create a greater radiation dose than current fuels. Therefore, hotter fuel may have to wait to cool to regulatory limits before handling or transporting, with cooling rates dependent on the thermal characteristics of the stored fuel, including the thermal conductivity of the ATFs and the canister configuration.

4.3.1.5 Subcriticality Considerations for ATF/AF

ATFs/AFs with higher density and enrichment may undergo increased criticality risk compared to light water reactor (LWR) fuels if the fuel relocates within a cask, or if a cask floods, either of which could occur during a transportation accident. These fuel properties may potentially lead to new storage, transportation, and disposal packaging designs (e.g., additional neutron absorbers).

4.3.2 Medium-Priority Storage and Transportation Challenges

4.3.2.1 Creep

Creep refers to the time-dependent deformation of a material that occurs if the hoop stress (driven either by internal rod pressure or pellet swelling) is large enough. For LWR fuel, creep concerns focused on cladding, so these concerns also apply to ATFs that are based on novel cladding materials. TRISO-based fuels could also potentially encounter creep issues depending on irradiation and thermal considerations (Jiang *et al*., 2021). Generally, this phenomenon is self-limiting since hoop stress will decrease when creep occurs and the volume (and thus pressure) of the rod decreases; hoop stress will also decrease as the temperature (and thus pressure) decreases over time. Creep presents a potential failure mechanism of the cladding since it results in thinning of the cladding wall. This thinning could lead to failure upon external loading or with increased rod internal pressure associated with higher burnup.

End of life rod internal pressures of the higher burnup fuels need to be measured and then modeled for the temperatures relevant to drying and early storage for ATF utilizing either zircaloy cladding coatings or alternate cladding materials. Cr, SiC, and FeCrAl will creep differently than zircaloy, so the limits before failure for these cladding types must be better understood.

4.3.2.2 Radiation damage

Radiation damage during reactor irradiation is a function of the fast neutron fluence and irradiation temperature, and this damage could occur either on ATF cladding or TRISO-based fuels. Radiation damage is known to increase the strength while reducing the ductility of the cladding. Annealing of radiation damage can decrease the hardness/strength and increase the ductility of the cladding, thus lessening the chance of failure from mechanical shock, but potentially facilitating additional cladding creep. Higher burnup is expected to result in increased radiation damage.

The material properties of irradiated ATF cladding, such as hardness and ductility, need to be determined to properly model the potential for cladding failure under various dry storage and transportation mechanisms.

Irradiation-induced cracking and debonding of the pyrolytic carbon layers has been identified as a potential failure mechanism for TRISO-based fuels (Hall et al., 2019c).

4.3.2.3 Consequences of confinement boundary failure

The primary confinement barrier during SNF dry storage is defined as the canister, and this definition could extend to the fuel container of a micro reactor. A failure, such as via stress corrosion cracking, could potentially release radionuclides in crud, oxide layers, fuel particulates, or fission gas. For sodium-bonded SNF, a confinement boundary failure could result in the exposure of sodium to moisture, resulting in the explosive generation of hydrogen gas. With TRISO-based fuels, failed fuel may be reactive with exposure to water and oxygen upon a confinement boundary failure. With molten salt fuels, fluoride gas release through a confinement boundary failure may be a concern.

The potential dose is higher in high burnup fuels and the form and size of fuel particulates is of high importance to modeling consequences of a failure.

4.3.2.4 Fuel fragmentation

Fragmentation of fuel may result in fuel relocation that could impact gas communication within a fuel rod for ATFs. This phenomenon could significantly affect release fractions upon breach of the cladding.

Finer particles are produced upon cladding breach as the burnup surpasses a threshold between 60 and 75 GWd/MTU. The behavior of ATF concepts, especially those at high burnup, should thus be ascertained. While Cr-doped fuels increase grain size and are expected to reduce fuel fragmentation, this has not been confirmed at high burnup.

Fuel kernel migration has been identified as a potential failure mechanism for TRISO-based fuels (Hall *et al*., 2019).

4.3.2.5 Fuel restructuring/swelling

Higher burnup results in increased fission gas production and will also result in increasing pressurization over time by generation of helium from alpha decay.

The Cr-doped pellets are expected to release less fission gas, but the retained gas could result in pellet swelling. This swelling may result in localized stress risers in the cladding, which in turn could facilitate other degradation mechanisms such as creep or stress corrosion cracking. This swelling could also occur in TRISO-based fuels (Jiang *et al*., 2021).

Failure of the SiC layer in TRISO-based fuels from internal fission gas pressure has been identified as a potential failure mechanism (Hall *et al*., 2019c).

4.3.2.6 Chemical attack

Pellet-cladding chemical interaction for uranium oxide fuels and zircaloy cladding is a fairly well-understood mechanism that has been reduced by changes to pellet geometry and the introduction of barrier fuels. The main danger associated with this interaction involves fission products in contact with the cladding that may facilitate cracking and breach of the cladding via pellet-cladding chemical interaction.

The compatibility of ATF cladding with the potential release of iodine, cesium, and cadmium from non-oxide fuels must be understood for long-term cladding performance.

Chemical attack of the SiC layer of TRISO-based fuels by fission products has been identified as a potential failure mechanism. This mechanism arises from the generation of palladium from the fuel kernel during irradiation that can form palladium silicides at localized reaction sites, which could compromise the integrity of the SiC (Hall et al., 2019c).

Fuel-cladding chemical interaction has been identified as a potential degradation mechanism for metallic fuels at high fuel burnup and high temperatures between 500-700 °C. The interaction could lead to cladding thinning and subsequent cracking (Hall et al., 2019c).

4.3.2.7 Fuel oxidation

Fuel oxidation in uranium oxide fuels can lead to rupturing of the cladding and result in fuel release and relocation. The kinetics of oxidation for ATFs under drying and storage conditions (resulting from retained water, water vapors, or radiolytically-generated oxidants) should be evaluated for long-term storage conditions. In addition, the stress necessary to initiate and propagate a crack in the various ATF cladding types should be studied.

For sodium-bonded metallic fuel, the reaction of the sodium with water/moisture in air would result in the generation of hydrogen gas and sodium hydroxide, so this phenomenon should be investigated further (Hall *et al*., 2019c).

4.3.2.8 Embrittlement

It has been suggested that Cr diffusion into the underlying zirconium-based cladding can result in embrittlement of the cladding. Potential chemical changes in ATF claddings, especially at the higher temperatures of drying and early storage, need to be examined. In addition, SiC claddings are known to be brittle (Honnold *et al*., 2021). There are some concerns with FeCrAl cladding embrittlement from irradiation at low temperatures – this may warrant further investigation.

4.3.3 Low-Priority Storage and Transportation Challenges

4.3.3.1 H_2 effects – cladding hydride reorientation

As cladding heats up when SNF is placed into a DCSS, hydrides may dissolve and, if the hoop stress is high enough, they can precipitate as radial hydrides as the cladding cools. If there are sufficient radial hydrides, the cladding response to mechanical loadings may change from an elastic response to brittle behavior as a function of temperature, based on the ductile-to-brittle transition.

However, ATF cladding designs are specifically meant to reduce oxidation and thus hydrogen generation and pickup, so these mechanisms are deemed to be of low importance.

4.3.3.2 Metal fatigue caused by temperature fluctuations

As protective layers around fuel such as cladding for ATFs and the outer pyrolytic carbon layer of TRISO-based fuels cools and warms up due to diurnal and seasonal temperature changes, the layer is subjected to thermal expansion stress and strain and associated metal fatigue. Gross cladding rupture can occur if the cladding is not resistant to this fatigue.

Because some ATF cladding concepts (such as SiC) are more brittle, there is reduced impact of fatigue compared to more ductile materials. For ATF, this reduces the priority of metal fatigue caused by temperature fluctuations, since other failure modes will be of higher concern/priority.

4.3.3.3 Phase change

The temperatures experienced in drying should be low enough that phase change of the ATF/AF components is not of concern.

4.3.3.4 Emissivity changes

While it will be important to understand the initial emissivity of the cladding to facilitate thermal modeling, past thermal modeling has demonstrated that the emissivity of the cladding surface has little effect on the temperatures.

4.3.4 Additional Uncategorized Storage and Transportation Challenges

4.3.4.1 Welded canister – fuel rod consolidation

There may be a need to reduce total waste package volume with ATF. If a consolidated rod canister is proposed as a solution, the canister design specifics will be used to maintain criticality control, meet shielding requirements, and provide adequate heat transfer are still to be determined.

The difference in loading process for a consolidated rod canister compared to LWR spent fuel should also be considered. For example, it remains to be determined whether a robotic rod gripping mechanism can be implemented directly on ATF without compromising fuel integrity. The robotic mechanism could employ special technology that modifies the geometry of the fuel for handling purposes if fuel integrity is a concern.

In terms of criticality, the criticality controls employed for LWR spent fuel rods need to be assessed to determine if these controls suffice for ATF.

4.4 Disposal Challenges

Many of the disposal challenges related to AR Fuels and ARWS were discussed previously in Section 3.2. Relative to Storage and Transportation, the time horizon for disposal is much longer (10^5 or 10^6 years) and as a result, discussion of the waste characteristics and associated disposal challenges are most easily done in tandem, as was done in Section 3.2. An analysis of disposal challenges can be organized via a Safety Case (NEA, 2013) , which includes elements for safety strategy, pre-closure assessments, and post-closure assessments. A key aspect of establishing the Safety Case is the Features, Events, and Processes (FEP) Screening. FEP Analysis and FEPs Screening are used interchangeably herein.

For post-closure safety assessment, analysis of FEPs, which includes FEP identification and FEP screening, provides a framework for a rigorous and systemic analysis of the safety of deep geologic repository during the post-closure period. The FEPs analysis is an exhaustive survey of all events that could occur post-closure, which includes rigorous documentation of the assumptions and bases for screening certain FEPs as critical drivers of overall repository performance. While many FEPs may pertain exclusively to the geologic aspects of the natural system – and therefore are not necessarily impacted by AR Fuels or ARWS characteristics – there are many FEPs that are impacted by the fuel types and their associated packaging, etc. In this section, the main challenges for disposal are reviewed by analyzing FEPs that are focused on fuel and packaging characteristics.

4.4.1 FEPs for Geologic Disposal of Used Advanced Fuels

A FEPs database contains a comprehensive tabulation of all FEPs potentially relevant to the evolution and performance of a complex system. The main purpose of a FEPs database is to aid in the methodical disposition of all potentially relevant FEPs so that there are no FEPs mistakenly overlooked.

One of the goals of the current report is to identify all processes and events that may be important to the disposal of Afs. Here, we build confidence that all such processes and events are considered by examining the Yucca Mountain Project FEPs database (SNL, 2008) and its application to YMP waste and its packaging.

The YMP FEPs potentially relevant to used fuels and waste package materials are identified in Sections 2.3.6 and 2.3.7 of the Safety Analysis Report (SAR) (DOE, 2008). Those FEPs are listed in Table 4-1 and Table 4-2 below. The first table lists the included (screened in) FEPs, and the second lists the excluded (screened out) FEPs.

A closer look at the 31 included YMP FEPs in Table 4-1 identifies the following potentially important processes and events for fuels and their packaging:

- General corrosion (uniform thinning, temperature dependence, patches)
- Stress corrosion cracking (initiation, propagation, stresses, patches)
- Localized corrosion (initiation, propagation, defect sites)
- Microbially influenced corrosion (relative humidity)
- Early failure (undetected defects, handling)
- Package physical form (strength, chemical behavior, dimensions, cladding, outer barrier)
- Radionuclide mass fractions in waste form and toxicity (radiation, chemical)
- Interaction between codisposed materials
- Waste form degradation processes (alteration, dissolution, radionuclide release)
- Chemical interaction with groundwater, package degradation products
- Radionuclide solubility and sorption (chemical environment)
- Colloids (intrinsic, pseudocolloids)
- Radioactive decay and ingrowth

Each of these processes and events is considered and discussed in Section 3.4.2 in relation to Afs.

The 25 excluded YMP FEPs in Table 4-2 were excluded in the YMP total system performance assessment due to low consequence or low probability (DOE, 2008). In general, the excluded FEPs pertain to the following processes and events:

- Alpha recoil
- Pyrophoricity
- Cladding corrosion and degradation
- Hydride cracking
- Internal corrosion of waste package materials prior to waste package breach
- Mechanical impact
- In-package and external criticality FEPs.

Many of the YMP FEP exclusions involved cladding. Because no credit was taken for cladding in the performance calculations, the excluded FEPs related to cladding would have had no consequence on the performance assessment results. Other FEPs, such as Copper Corrosion, were excluded because they were not applicable. A particular FEP excluded due to low probability was Hydride Cracking of Waste Packages. It was excluded because it could be shown that the waste package materials were highly resistant to hydrogen-induced cracking. Whether these same FEPs would be excluded or included for AR SNF remains to be seen in FEP analyses for the specific AR SNF in generic disposal concepts.

The in-package and external Criticality FEPs were excluded for the YMP on the basis of low probability for a variety of SNF types (e.g., oxides, metals, TRISO, alloys) with a range of enrichments (ranging from natural uranium to greater than 90 wt.% U-235) and fissile materials (U-233, U-235, and Pu-239). The low probability determination for the in-package criticality FEPs was based on low probability of package failure (resulting in water intrusion) in combination with disposal canister design and SNF loading criteria. The canister designs for various SNF types included corrosion resistant neutron absorber plates and/or fillers with embedded low-solubility neutron absorbers. The SNF loading criteria included limits on number of fuel elements, enrichment/burnup combinations, and other fuel-specific characteristics. The external criticality FEPs were excluded based on the near-field design and tuff characteristics, including porosity, fractures, and lithophysae sizes and distributions. It was concluded that insufficient fissile material could be released from the waste package and accumulate in the near-field or the far-field to result in criticality potential based on the 10,000-year regulatory period. Several factors could affect the criticality FEPs screening evaluations for ARF and ARWS including packaging, loading, disposal geology, regulatory period, and licensing strategy. These factors determine whether the criticality FEPs for the disposal of ARF would be excluded based on low probability, low consequence, or included in a repository performance assessment.

A similar list of potentially important processes and events for used fuel would likely be produced using other well-established FEP databases. For added rigor, additional examination of the NEA FEPs data could be done to see if there are any additional processes or events relevant to waste forms that are not included in the YMP FEPs database.

Regardless, this FEPs exercise has provided a long list of processes and events potentially important to Afs. The results build confidence that all potentially important FEPs are considered in this report. Further details on these FEPs and the processes and events they include are available in the references cited.

Table 4-1 Fuel-related YMP FEPs *included in* YMP performance assessment calculations

2.1.01.01.0A	Waste inventory (screening for radiotoxicity and chemical toxicity)
2.1.01.02.0B	Interactions between co-disposed waste (in-package chemistry)
2.1.01.03.0A	Heterogeneity of waste inventory (individual waste package types)
2.1.01.04.0A	Repository-scale spatial heterogeneity of emplaced waste
2.1.02.01.0A	DSNF degradation (alteration, dissolution, and radionuclide release)
2.1.02.02.0A	CSNF degradation (alteration, dissolution, and radionuclide release)
2.1.02.07.0A	Radionuclide release from gap and grain boundaries (IRF)
2.1.02.09.0A	Chemical effects of void space in waste package (IPC)
2.1.02.12.0A	Degradation of cladding prior to disposal
2.1.02.23.0A	Cladding unzipping
2.1.02.25.0B	Naval SNF cladding
2.1.02.28.0A	Grouping of DSNF waste types into categories (instant degradation except naval SNF represented as CSNF)
2.1.03.01.0A	General corrosion of waste packages (temperature-dependent, patches)
2.1.03.02.0A	Stress corrosion cracking of waste packages (crack initiation, patches)
2.1.03.03.0A	Localized corrosion of waste packages
2.1.03.05.0A	Microbially influenced corrosion of waste packages (RH)
2.1.03.08.0A	Early failure of waste packages
2.1.03.11.0A	Physical form of waste package and drip shield (strength, chemistry)
2.1.09.01.0B	Chemical characteristics of water in waste package (IPC)
2.1.09.02.0A	Chemical interaction with corrosion products (EBS transport abstraction)
2.1.09.04.0A	Radionuclide solubility, solubility limits, and speciation in the waste form and EBS
2.1.09.05.0A	2.1.09.05.0A Sorption of dissolved radionuclides in EBS (onto corrosion products)
2.1.09.06.0A	Reduction-oxidation potential in waste package (atmospheric conditions)
2.1.09.07.0A	Reaction kinetics in waste package (IPC)
2.1.09.17.0A	Formation of pseudo-colloids (corrosion product) in EBS
2.1.09.23.0A	Stability of colloids in EBS
2.1.09.25.0A	Formation of colloids (waste-form) by co-precipitation in EBS
2.1.11.08.0A	Thermal effects on chemistry and microbial activity in the EBS
2.2.07.06.0B	Long-term release of radionuclides from the repository
3.1.01.01.0A	Radioactive decay and ingrowth
2.1.01.01.0A	Waste inventory (screening for radiotoxicity and chemical toxicity)

Table 4-2 Fuel-related YMP FEPs *excluded from* YMP performance assessment calculations.

2.1.01.02.0A	Interactions Between Co-Located Waste
2.1.02.04.0A	Alpha Recoil Enhances Dissolution
2.1.02.08.0A	Pyrophoricity from DSNF
2.1.02.10.0A	Organic/Cellulosic Materials in Waste
2.1.02.11.0A	Degradation of Cladding from Waterlogged Rods
2.1.02.13.0A	General Corrosion of Cladding
2.1.02.14.0A	Microbially Influenced Corrosion (MIC) of Cladding
2.1.02.15.0A	Localized (Radiolysis Enhanced) Corrosion of Cladding
2.1.02.16.0A	Localized (Pitting) Corrosion of Cladding
2.1.02.17.0A	Localized (Crevice) Corrosion of Cladding
2.1.02.18.0A	Enhanced Corrosion of Cladding from Dissolved Silica
2.1.02.19.0A	Creep Rupture of Cladding
2.1.02.20.0A	Internal Pressurization of Cladding
2.1.02.21.0A	Stress Corrosion Cracking (SCC) of Cladding
2.1.02.22.0A	Hydride Cracking of Cladding
2.1.02.24.0A	Mechanical Impact on Cladding
2.1.02.25.0A	DSNF Cladding
2.1.02.26.0A	Diffusion- Controlled Cavity Growth in Cladding
2.1.02.27.0A	Localized (Fluoride Enhanced) Corrosion of Cladding
2.1.02.29.0A	Flammable Gas Generation from DSNF
2.1.03.04.0A	Hydride Cracking of Waste Packages
2.1.03.06.0A	Internal Corrosion of Waste Packages Prior to Breach
2.1.03.07.0A	Mechanical Impact on Waste Package
2.1.03.09.0A	Copper Corrosion in EBS
2.1.03.10.0A	Advection of Liquids and Solids Through Cracks in the Waste Package
2.1.14.15.0A	In-package criticality (intact configuration)
2.1.14.16.0A	In-package criticality (degraded configurations)
2.1.14.18.0A	In-package criticality resulting from a seismic event (intact configuration)
2.1.14.19.0A	In-package criticality resulting from a seismic event (degraded configurations)
2.1.14.21.0A	In-package criticality resulting from rockfall (intact configuration)
2.1.14.22.0A	In-package criticality resulting from rockfall (degraded configurations)
2.1.14.24.0A	In-package criticality resulting from an igneous event (intact configuration)
2.1.14.25.0A	In-package criticality resulting from an igneous event (degraded configurations)
2.2.14.09.0A	Far-field criticality
2.2.14.10.0A	Far-field criticality resulting from a seismic event
2.2.14.11.0A	Far-field criticality resulting from rockfall
2.2.14.12.0A	Far-field criticality resulting from an igneous event
2.1.14.17.0A	Near-field criticality
2.1.14.20.0A	Near-field criticality resulting from a seismic event
2.1.14.23.0A	Near-field criticality resulting from rockfall
2.1.14.26.0A	Near-field criticality resulting from an igneous event

5 DISPOSAL PATHWAYS

5.1 DOE Managed Spent Nuclear Fuels as Precedent Disposal Pathways

There are several precedent disposal pathways that are available as a basis for comparison for analyzing the feasibility of AR Fuels and ARWS disposal pathways. The 2014 Disposal Options Reports (SNL, 2014) consolidates the ~40-50 waste types into 10 waste groups. A waste group represents a collection of waste types with enough similarity in waste characteristics such that the disposal options for the Group can be analyzed collectively.

Table 5-1. Precedent Waste Groups from the 2014 Disposal Options Report (from SNL 2014)

Waste group	Description
WG1	All commercial SNF packaged in purpose-built disposal containers
WG2	All commercial SNF packaged in dual-purpose canisters of existing design
WG3	All vitrified HLW (all types of HLW glass, existing and projected, canistered)
WG4	Other engineered waste forms
WG5	Metallic and non-oxide DOE spent fuels
WG6	Sodium-bonded fuels (driver and blanket), direct disposed[1]
WG7	DOE oxide fuels
WG8	Salt, granular solids, and powders
WG9	Coated-particle spent fuel
WG10	Naval fuel

[1]Note: it was concluded that insufficient data exist to evaluate direct disposal of sodium-bonded fuels.

Of the 10 waste groups (WGs), WG4 through WG9 potentially have the greatest relevance for AR Fuels and ARWS. The list of potential precedents can be parsed as follows:

- WG1/WG2 covers all commercial LWR SNF and could provide precedent for accident tolerant fuels (ATFs)
- WG4 – Electrometallurgical Treatment (EMT) wastes (used to treat sodium-bonded metallic fuels)
- WG5 Metallic Fuels and WG6 Sodium Bonded Metallic Fuels provide potential precedents for Metallic Fuels broadly, covering Metallic fuels (WG5) -which excludes Sodium bonded - and Sodium Bonded Metallic Fuels (WG6).
- WG7 DOE Oxide Fuels provides potential precedent for AR Fuels based on LWR fuel.
- WG8 Salt, Granular Solids, and Powders provides potential precedent for MSR waste streams
- WG9 Coated-Particle Spent Fuels provides potential precedent for TRISO fuels.

5.2 Advanced Reactor Spent Nuclear Fuel Disposal Strategies

This chapter compares AR SNF Types with existing waste groups (as outlined above) to determine the feasibility. The Evaluation Criteria are shown in Table 5-2 below.

Table 5-2. Evaluation Criteria used to Assess disposal Strategies for AR Fuels and ARWSs (from SNL, 2014).

Evaluation Criteria	Metrics
Disposal Option Performance	Likely to comply with long-term standards?
Confidence in Expected Performance Bases	Additional EBS components needed above baseline for each design concept Robustness of information bases; simplicity vs. complexity; knowledge gaps
Operational Feasibility	Ease in ensuring worker health and safety at all stages Special physical considerations at any stages based on physical characteristics
Secondary Waste Generation	Amount of low-level waste generated during handling and treatment Amount of mixed waste generated
Technical Readiness	Status of waste form technologies Status of transportation and handling systems Status of disposal technologies
Safeguards and Security	National security implementation difficulty Radiological dispersion device prevention implementation difficulty

5.2.1 TRISO fuels

TRISO fuels were used in the Peach Bottom Reactor and the Fort St. Vrain reactor, both of which are of the prismatic block design. These reactors are no longer operational in the U.S. TRISO fuels are classified WG9- Coated-particles Spent Fuel (SNL, 2014). Table 5-3 shows the summary evaluation for the four repositories under consideration in the U.S. Program – Salt, Crystalline, Clay/Shale, and Deep Borehole.

Table 5-3. Summary of Coated Particle Spent Fuel (from SNL, 2014)

Disposal Concept	Disposal Option Performance		Confidence in Expected Performance Bases		Operational Feasibility		Secondary Waste Generation		Technical Readiness		Safeguards and Security	
Salt	✓		✓		✓		✓		✓		✓	
Crystalline	✓		✓		✓		✓		✓		✓	
Clay/Shale	✓		✓		✓		✓		✓		✓	
Deep Borehole	✓	NA	○	NA	○	✕	✓	NA	○	NA	✓	NA

Note: Split scores indicate that size constraints preclude disposal of some, but not all, waste forms in this group.

Legend:	✓ Strong	○ Moderate	✹ Weak/Uncertain	✕ Not Feasible	NA Not analyzed

It is clear from Table 5-3 that TRISO fuels have strong metrics for all five Evaluation Criteria for Salt, Crystalline, and Clay/Shale Repository concepts. There are some operational feasibility issues for deep borehole, which are related to the fact that some of this fuel type have graphitic core diameters that are

too large for deep borehole disposal. Prior experience with TRISO, and a detailed discussion of the Evaluation Criteria appear in the subsequent sections below.

5.2.1.1 Storage and Transportation

TRISO fuel from the Fort St. Vrain reactor has been transported via truck in the TN-FSV transportation cask. Operation of the high-temperature gas-cooled reactor generated ~ 23.35 MTHM of SNF; ~ 15 MTHM of that SNF is currently stored in sealed carbon steel containers in an air-cooled DOE-owned and NRC-licensed storage facility in Colorado while the remaining ~8 MTHM is stored at Idaho National Laboratory (INL) in carbon steel canisters at storage area CPP-603. TRISO fuel from operation of the Peach Bottom Unit 1 is currently stored at INL in carbon steel-lined, double O-ring sealed aluminum canisters at storage area CPP-749. (NWTRB, 2017; NWTRB, 2020).

The NWTRB identified some technical issues related to the storage of the Fort St. Vrain TRISO fuel (NWTRB, 2017; NASEM, 2022). These issues included aging management concerns associated with the silicon carbide layer of the TRISO particle during extended storage, due to differing degradation and corrosion processes of carbide-based fuels compared to oxide-based fuels. The concerns include gas generation, mechanisms of waste form degradation, mechanisms of dissolution and precipitation of specific radionuclides, radionuclide solubility limits, and the sorption/desorption of radionuclides on degraded waste package materials (NASEM, 2022).

For transportation, TRISO fuels provide a robust, multi-layered confinement boundary that is likely to survive both normal conditions of transport and hypothetical accident conditions. However, the higher fuel enrichments from the use of HALEU in TRISO fuels would entail criticality analyses to ensure regulatory compliance, while the higher burnups expected from reactors using TRISO fuels would have thermal analyses to ensure that the 50 °C 10 CFR 71 transportation cask surface temperature limit is not reached (NASEM, 2022).

In the case of TRISO pebble deployment in a salt-cooled advanced reactor, it is possible that salt contamination (or traces thereof) may remain on the surface of TRISO pebbles. In this case, research and analysis would address: 1) if appreciable and/or consequential amounts of salt residue persists on TRISO fuel pebbles, and 2) potential implications of such salt residue on the surface of TRISO fuel pebbles with respect to Storage and Transportation analyses. For example, off-gassing from salt residue could affect fuel dispositioning, including considerations of canister integrity due to both the mechanical and chemical effects of residual salt off-gases potentially reacting on the interior of the waste container.

5.2.1.2 Disposal

Disposal of TRISO fuel was included in the disposal plan for Yucca Mountain (DOE, 2010). Spent TRISO fuel particles are expected to retain radionuclides better than LWR spent fuel in a geologic repository (Hall et al., 2019a); thus, this is a waste form that is suitable for disposal and does not need to be treated (other than packaging in an appropriate disposal container or separation from the graphite block, if desired) prior to disposal. If the reactor design included a graphite prismatic block, as did the Fort St. Vrain reactor, this block would either be disposed of with the fuel or the fuel would be separated from the block, followed by deep geologic disposal of the fuel and disposal of the block as low-level waste. Disposal of the block along with the fuel increases the volume of waste to be disposed with a corresponding decrease in the heat generated per volume of waste.

This waste generates ^{14}C and ^{3}H from irradiating graphite and lithium, which is an impurity in graphite (Kitcher, 2020a); these isotopes would need to be considered in the performance assessment that would be conducted to assess compliance with disposal regulatory requirements. In addition, when carbide is exposed to water, flammable gases are generated (SNL, 2008). While generation of these gases was

considered inconsequential in terms of post-closure repository performance for the unsaturated proposed Yucca Mountain repository, gas generation might be more significant in a repository in saturated geologic media where repository pressurization is possible. As mentioned in the previous subsection (5.2.1.1), TRISO fuel used in a salt-cooled reactor could have residual salt on/in the pebble surface, and the effects of off-gassing and/or reactivity of this salt could also have implications that effect disposal assessment/performance.

Finally, TRISO fuels tend to be more highly enriched than typical LWR fuels; therefore, additional measures may need to be taken to ensure disposal criticality control (DOE, 2010). Because of: (1) the near-optimally moderated graphite blocks, (2) the relatively smaller disposal canisters for Yucca Mountain that were designed to dispose of 5 vertically stacked blocks, and (3) corrosion resistant graphite and TRISO particles, neutron absorbers were not included for disposal of the Fort St. Vrain TRISO spent fuel.

In comparison to other fuel cycles discussed here, the TRISO-type fuel cycle appears to be the WF most suited for direct disposal in generic mined repositories and deep boreholes, without the need for complex materials treatment of other fuel cycles. A notable exception being the previously discussed case in which TRISO fuel reacted within salt coolant may have appreciable salt residue on TRISO fuel surface. Also notable is the challenge to determining the ultimate isotopic composition of TRISO pebbles from pebble bed reactors due to their varyiable/uncertain burnup histories, and initial enrichment levels. This may benefit from development of newer technologies such as radioactive pebble identification, anticipatory pebble monitoring, estimation of fuel depletion, and additional Non-Destructive Analysis methods (NASEM 2023, pp. 185).

As summarized in Ch.3, since the TRISO particles are bonded together into graphite pebbles/compacts, recoveringindividual spent TRISO particles before disposal would entail considerable effort in terms of fuel treatment. Direct disposal of whole graphite prismatic fuel elements/blocks containing the compacts and TRISO pebbles containing spent TRISO pellets is considered the preferred option, owing to its lower cost, ease of implementation, and limited safeguards requirements and proliferation risk (Lotts *et al*, 1992; IAEA, 2013; Hall, 2019a).

Although there is currently no experience with actual disposal of spent coated fuel particles (Hall, 2019b), previous experience exists for the long-term storage of TRISO-type particles or buffered isotropic coated particles or monopyrolytic carbon coated particles in graphite (see, e.g., Fort Saint Vrain (FSV) HTGR TRISO fuel & Peach bottom fuel particles).

Several evaluation criteria and associated metrics (see Table 5-2) are discussed for TRISO-type disposal:

Disposal Option Performance:

(1). *Likely to comply with long-term standards*: Probably. Older TRISO fuels had some problems with early fuel failure, but newer TRISO fuels, such as those expected to be produced in the future would probably be long-lived in any environment because of the layer of SiC, but this would be an area for potential research. TRISO fuel pebbles used as fuel in an MSR may take on elements from the salt coolant that diffuse into the graphite matrix.

(2). *Identification of key attributes of the disposal system:* With an average burnup of ~120 GWd/t, these are very high burnup and thus are presumably very hot. If disposed of directly, the thermal response of the disposal system would be important. If disposed of directly in a deep borehole, the fuel compacts would have to be removed from the graphite blocks, which may be too large to fit down a borehole.

Confidence in Expected Performance Bases:

(1). *Additional EBS components needed above baseline for each design concept*: None.

(2). *Robustness of information bases; simplicity vs. complexity; knowledge gaps*: These have not been studied in terms of their long-term performance. How stable are they? How stable is the graphite at high temperature? Is it possible for them to go critical in a repository? Graphite is the moderator in the HTGR, so introduction of water may not significantly increase reactivity, which was the conclusion of the disposal plan for Yucca Mountain (DOE, 2010), but this should be confirmed based on the design of the new fuel and packaging (e.g., disposal with or without graphite blocks). Reactor performance data and analysis will be needed to fully characterize TRISO fuel used in a pebble-bed type reactor. The information base for borehole disposal is not as substantial as that for other disposal options.

Operational Feasibility:

(1). *Ease in ensuring worker health and safety at all stages*: For disposal in a deep borehole, TRISO particle fuels in graphite fuel blocks may have to be removed from the graphite blocks, as the graphite blocks are too big to fit in a deep borehole, thus potentially increasing the dose to exposed workers performing the extractions. For other disposal concepts/options, this dimensional limitation on canister diameter does not apply. This fuel is high burnup, so it would produce a significant radiation field.

(2). *Special physical considerations at any stages based on physical characteristics*: The compacts may have to be removed from the graphite blocks for disposal in a deep borehole.

Secondary Waste Generation:

(1). *Amount of LLW generated during handling and treatment*: Probably not a candidate for reprocessing. If the compacts were removed from the graphite blocks to facilitate disposal of the compacts in a deep borehole, the graphite blocks would represent a waste stream to undergo classification (e.g., Class A, B, C, or GTCC LLW) and disposal.

(2). *Amount of mixed waste generated*: Analysis of graphite blocks needed.

Technical Readiness:

(1). *Status of needed waste form generation technology*: Not applicable, unless removing compacts from graphite blocks.

(2). *Status of transportation and handling systems*: Fuel of this type is being stored at the Fort Saint Vrain storage facility in Colorado, which is licensed by the NRC. The TN FSV transport cask is licensed to transport Fort St. Vrain spent fuel. Peach Bottom TRISO spent fuel is being stored at an NRC-licensed ISFSI at the INL; the TN FXV transport cask is licensed to transport the Peach Bottom irradiated fuel.

(3). *Status of disposal technologies*: All disposal concepts are ready for this spent fuel, except possibly deep borehole.

Safeguards and Security:

(1). *National security implementation difficulty (fissile content)*: This spent fuel has the potential to have a high fissile content, likely with increased material control and accounting (MC&A) and safeguards.

(2). *Radiological dispersion device prevention implementation difficulty*: Individual TRISO particles would be easily dispersible; but obtaining individual TRISO particles could be challenging.

5.2.2 Metallic Fuels

Metallic Fuels (U-TRU-Zr alloy) are typically used in sodium cooled, fast reactors. The metallic nature of these fuels has two important consequences, namely, the fuel will swell upon exposure to the heat of fission and produce gaseous by-products. These operational characteristics of the fuel mean that fuel design needs to be taken into account for the build-up of gases via a plenum, as well as be engineered with an intermediary material between the fuel and the cladding that can account for the fuel swelling while also providing good heat transfer characteristics so that fuel temperature can be managed appropriately (FRWG, 2018). The N-Reactor at the Hanford Site produced a significant amount of metallic fuel, while the EBR-II Reactor incorporated a "sodium-bond" between the fuel and cladding, producing a distinct metallic fuel sub-type – Sodium-Bonded Fuel. Gen IV Sodium Fast Cooled Reactors are likely to draw upon EBR-II's demonstration of technical feasibility of sodium-bonded fuel, until a more developed AF concept arises. Sodium-bonded fuels are not amenable to direct disposal and should undergo treatment to remove pyrophoric metallic sodium from other fuel constituents. Advanced metallic fuels may be developed without the sodium bond, and as such, may have a direct disposal pathway depending on their total amount, reactivity, and waste form lifetime.

Metallic fuels have a precedent disposal pathway via WG5 – Metallic Fuels, while Sodium Bonded Fuels fall into WG6 – Sodium Bonded Fuels. Table 5-4 (SNL 2014) shows the WG5 summary evaluation for the four repositories under consideration in the U.S. Program.

Table 5-4. Summary of Metallic Spent Fuel (from SNL, 2014)

Disposal Concept	Disposal Option Performance		Confidence in Expected Performance Bases		Operational Feasibility		Secondary Waste Generation		Technical Readiness		Safeguards and Security	
Salt	✓		✓		○		✓		○		✓	
Crystalline	✓		○		○		✓		○		✓	
Clay/Shale	✓		✓		○		✓		○		✓	
Deep Borehole	✓	NA	○	NA	✓	✕	✓	NA	○	NA	✓	NA

Note: Split scores indicate that size constraints preclude disposal of some, but not all, waste forms in this group.

Legend:	✓ Strong	○ Moderate	✹ Weak/Uncertain	✕ Not Feasible	NA Not analyzed

It is worth noting that Metallic Fuels (WG5), relative to oxide fuels typical of LWRs, are likely to be more reactive and hence have a larger relative source term contribution. In the SNL (2014) study, the WG5 was a small fraction (<3%) of the total inventory being considered for disposal. If WG5 fuels were to comprise a large fraction of the total inventory, further analyses and/or performance assessment may be necessary to understand the effect of such an enhanced source term and make a determination if an alternative to direct disposal is optimal and/or feasible.

There are several types of metallic fuels that have been produced, processed, stored, and/or transported in the U.S:

- N Reactor fuel is a zircaloy-clad low-enriched uranium metal fuel. It is currently stored at the Hanford site in 24-inch multi-canister overpacks (MCOs) inside the Canister Storage Building (CSB). The canisters have only been moved on site. The ability to transport the MCOs offsite under 10 CFR 71 has not been fully evaluated.
- EBR-II fuel is a sodium-bonded metal fuel composed of either U-5Fs (Fs = 2.46Mo + 1.96Ru + 0.28Rh + 0.19Pd + 0.085Zr + 0.02(Nb+Ta) + 0.054Si), U-10Zr, or U-20Pu-10Zr with stainless steel cladding (Fredrickson, 2022; Pope, 2017). EBR-II spent fuel currently resides at the Idaho Nuclear Technology and Engineering Center (INTEC) in water storage pools, Radiological Scrap and Waste Facility (RSWF) in underground silos, and within the Fuel Cycle Facility (FCF) and Hot Fuel Examination Facility (HFEF) hot cells (Fredrickson, 2022).
- Single-Pass Reactor fuel is an aluminum clad low-enriched uranium metal fuel. It is currently stored in canisters similar to the N Reactor MCOs in the CSB.
- Enrico Fermi fuel is a uranium-molybdenum alloy fuel enriched to approximately 26 wt% U-235. The Fermi fuel that is packaged in small cans and stored in a pool at the Idaho site is metallurgically bonded to a zirconium tube that serves as cladding, resulting in no gap between cladding and fuel.
- Advanced test reactor (ATR) fuel is an aluminum clad highly enriched uranium-aluminum alloy fuel. Although ATR makes up a significant fraction of uranium-aluminum alloy fuel inventory, there is wide diversity in this category. The fuel has recently been moved from wet to dry storage at the Idaho site. ATR and several other uranium-aluminum alloy fuels have been transported primarily to the Savannah River Site for storage and reprocessing.

5.2.2.1 Sodium-Bonded Metallic Fuels

While WG5 (Metallic Fuels) has been previously analyzed for disposal pathways, WG6 (Sodium-bonded Metallic Fuels) direct disposal pathways have not been previously analyzed due to the lack of information to perform an analysis. Previous analyses have assumed Electrometallurgical Treatment (EMT) that would produce WG4 and/or WG8 waste forms (SNL, 2014b).

Sodium-bonded fuel is used in sodium-cooled fast (spectrum) reactors and uses either uranium or plutonium metal as the fuel material. In the U.S., sodium-bonded metallic spent fuel was generated by the Experimental Breeder Reactor–II (EBR-II) in Idaho, the Fast Flux Test Facility at Hanford, the Sodium Reactor Experiment, and the Fermi-1 unit in Michigan (Kitcher, 2020b). None of these reactors are still in operation, and not all of the fuel from these reactors is sodium-bonded. Disposal of Fast Flux Test Facility and Fermi SNF that is not sodium-bonded (which comprises the majority of the fuels) was included in the disposal plan for Yucca Mountain (DOE, 2010); these metal fuels without sodium have also been transported by the DOE and are currently being stored by the DOE. These fuels do present some potential challenges with respect to the BENFC and were discussed in the previous section.

5.2.2.1.1 STORAGE AND TRANSPORTATION

Sodium-bonded spent fuel has been transported both between states and within INL. The T-3 canister/cask system was used to transport sodium-bonded spent fuel generated by the Fast Flux Test Reactor from the Hanford Site in Washington to INL, and the PB-1 shipping cask was used to transport sodium-bonded spent fuel generated by the Fermi-1 reactor from Michigan to INL (PRDC 1975; Hall et al., 2019b). Sodium-bonded spent fuel generated by the EBR-II reactor was transported within INL. Three casks were certified for transporting sodium-bonded spent metal fuel: Model MLI-1/2, T-3 Model, and

NAC-LWT. Only the NAC-LWT system currently has an unexpired certificate of compliance (Hall et al., 2019b).

All of the sodium-bonded spent fuel is stored at INL and is being moved (or has already been moved) from wet storage to dry storage at the Materials and Fuels Complex, which consists of the Hot Fuel Examination Facility and the Radioactive Scrap and Waste Facility (NWTRB, 2017). None of these storage facilities is licensed by the NRC. The casks previously used at Hanford consisted of an inner double-encapsulated steel container that contains the fuel and an outer steel container that holds lead shielding and the inner container. The storage systems used for this fuel at INL have not been described in publicly available documents (NWTRB, 2017).

During the period of time that the sodium-bonded spent fuel was stored under water at INL, some of the stainless-steel storage containers holding spent sodium-bonded fuel leaked because they had not been sealed properly (Pahl, 2000). One container was chosen for detailed destructive analysis; in that container the headspace gas was more than 99% hydrogen, the fuel had ruptured, and fuel oxide sludge filled the bottom of the container. The metallic sodium had reacted with the water and formed a concentrated caustic solution of NaOH. In addition, some sodium-bonded driver fuel in dry storage inside seal-welded containers at INL reacted with moisture internal to the storage canisters, evolving hydrogen and splitting the cladding (DOE, 2006).

The primary storage challenges for sodium-bonded spent fuel are focused on the prevention of the reaction of sodium metal with water and/or water vapor. As such, the fuel cladding as well as the storage package should be resistant to moisture degradation. The storage environment needs to be taken into account, as well, so as to protect the sodium from exposure to moisture. For long-term storage, the sodium could be removed from the fuel via some form of treatment (such as EMT) to eliminate this as a potential further degradation mechanism for the storage cask from internal moisture to prevent potential exposure of the cask interior to external moisture. Criticality and thermal analyses would be conducted for sodium-bonded fuels from advanced reactors that plan to use HALEU (NASEM, 2022).

The primary transportation challenges for sodium-bonded spent fuel are similarly focused on keeping sodium metal from reacting with water. Analyses for transportation packages would be carried out to ensure sufficient confinement and shielding of the fuel during both normal conditions of transport and hypothetical accident conditions. If HALEU is used, criticality and thermal analyses would need be done to ensure that the higher enrichments still meet regulatory requirements for transportation (NASEM, 2022).

5.2.2.1.2 DISPOSAL

Sodium-bonded spent fuel contains metallic sodium in the gap between the fuel pellets and the cladding. Metallic sodium is highly reactive with water, making sodium-bonded spent fuel unsuitable for direct disposal. Thus, this fuel must be treated prior to disposal. Note that blanket fuel does not experience as much irradiation during reactor operation than the driver fuel, so the treatment approaches for blanket fuel may differ from those for driver fuel (Hall et al., 2019b). Several different treatment options have been identified: (1) EMT, (2) melt and dilute, (3) distillation of blanket fuel (i.e., melt, drain, evaporate, carbonate (MEDEC)), and (4) aqueous processing (SNL, 2014b). In 2000, the DOE selected the EMT process to treat its inventory of sodium-bonded spent fuel from the Experimental Breeder Reactor II (EBR II) and from the Fast Flux Test Facility (DOE, 2000). As of 2017, 4.82 MTHM of sodium-bonded SNF from EBR-II, both driver and blanket, had been reprocessed via EMT (Rechard et al., 2017). A treatment process for blanket sodium-bonded spent fuel from Fermi-1 has not yet been selected.

The EMT process, which is carried out in the Fuel Conditioning Facility at INL, produces a uranium product, a metallic waste, and a high-level salt radioactive waste (SNL, 2014b). The uranium product is not a waste but is stored for later use. The metallic waste is suitable for disposal as-is and does not need further treatment, other than packaging for disposal. The chloride-rich salt waste would dissolve quickly in the presence of water, making radionuclides available for transport away from the waste package and forming a chloride-rich electrolyte that is highly corrosive to metallic barriers or other waste forms in the disposal system (Hall et al., 2019a). Thus, these chloride-rich salts might be suitable for disposal in a salt repository but present challenges for disposal in non-salt formations and so may need to be further treated prior to disposal. Vitrification of the salt waste is not feasible because of its high chloride content. The current plan is to first occlude the waste salt within a zeolite matrix and then microencapsulate the zeolite in a borosilicate glass, thus producing a glass-bonded sodalite material referred to as the ceramic waste form (CWF) that is suitable for disposal (SNL, 2014b). Dissolution rates for the CWF are generally lower than dissolution rates for HLW glass (Hall et al., 2019a). To date, the salt waste produced by the EMT process has not been treated to produce a CWF and is still in the electrorefiners that produced it (Frederickson et al., 2022).

5.2.2.2 Sodium-Cooled Fast Reactor Metal Fuel Cycle (Pyroprocessing Recycle)

Due to the expected operating conditions for sodium-cooled fast reactor (SFRs), some fissile material present in spent fuel from other reactors may be consumed in SFRs, thus reducing the total amount of fissionable waste. However, Sodium-bonded fuels will likely undergo treatment of the spent fuel after removal from the sodium coolant and before pyroprocessing, due to potential reaction between elemental sodium and air or water. To date, direct disposal of these fuels has not been considered in safety assessments for deep geologic disposal. Because of the reactive nature of these spent fuel waste forms and the challenges that they present, they represent a challenging disposal category.

Several evaluation criteria and associated metrics are listed below for disposal of sodium-bonded spent fuel from to the SFR metallic fuel cycle:

Disposal Option Performance:

(1). *Likely to comply with long-term standards?*: This waste form is not compatible with any repository environment because of the pyrophoric nature of the fuel in its current form. The question then becomes whether the waste forms produced by treatment of the sodium-bonded spent fuel are likely to comply with long-term disposal standards. Thermal, chemical, physical, and packaging characteristics would come into play.

(2). *Identification of key attributes of the disposal system:* This SNF is also hotter than typical LWR UOX fuel, but probably also not disposable as-is. The question then becomes the thermal, chemical, and physical characteristics of the waste forms produced by treatment of the sodium-bonded spent fuel.

Confidence in Expected Performance Bases:

(1). *Additional EBS components needed above baseline for each design concept*: Depends on the waste forms produced during treatment.

(2). *Robustness of information bases; simplicity vs. complexity; knowledge gaps*: This metric would apply to waste forms produced by reprocessing this fuel. Processing of fuel to remove sodium bond

Operational Feasibility:

(1). *Ease in ensuring worker health and safety at all stages*: Sodium-bonded fuel will have to be reprocessed, making it more difficult to avoid potential increases in worker dose. The composition of the final waste forms would be a factor.

(2). *Special physical considerations at any stages based on physical characteristics*: The compacts may have to be removed from the graphite blocks for disposal in a deep borehole.

Secondary Waste Generation:

(1). *Amount of LLW generated during handling and treatment*: Reprocessing would probably generate LLW, and, depending on the reprocessing method used, may generate a uranium product, which is not waste but must be managed. These waste forms would await classification.

(2). *Amount of mixed waste generated*: Analysis of proposed waste form(s) needed. If the waste treatment process creates a uranium product, management of this product would be a consideration.

Technical Readiness:

(1). *Status of needed waste form generation technology*: Pyroprocessing of sodium-bonded spent fuel has been developed (DOE, 2000). Sodium-free annular metal fuel does not undergo this first separation. The status of other potential reprocessing techniques is unknown.

(2). *Status of transportation and handling systems*: Currently stored by the DOE, not in an NRC-licensed storage facility.

(3). *Status of disposal technologies*: Sodium-bonded waste is not disposable as-is. All disposal concepts may be suitable for the waste forms that might be produced from treatment of this waste, except for the outstanding feasibility R&D on deep borehole disposal.

Safeguards and Security:

(1). *National security implementation difficulty (fissile content)*: Fissile content may be high depending on whether this fuel was used in the once-through cycle or in the "continuous recycling" fuel cycle and on which reprocessing approach was used (e.g., fissile material extracted from the other constituents). High fissile content waste forms would likely have increased MC&A and safeguards.

(2). *Radiological dispersion device prevention implementation difficulty*: Exposing sodium-bonded fuel to water could create a radiological dispersion device; the opportunity to do this could be non-existent. This is also a metric for waste forms produced by reprocessing this waste.

5.2.2.3 Sodium-Cooled Fast Reactor Metal Fuel Cycle (Once-Through, Treated)

The disposal evaluation criteria and associated metrics for the sodium-bonded spent fuel for SFR metal fuel cycle, once-through, treated, are likely to exhibit a high degree of similarity with that of "Sodium-Cooled Fast Reactor Metal Fuel Cycle (Pyroprocessing Recycle)" discussed above, except that the non-pyroprocessing treated waste may likely be classified as HLW.

5.2.3 Molten Salt Reactor Waste Streams

Molten Salt Reactor Waste Streams could have disposal precedents in either WG 4 (other waste forms, such as glass or ceramic waste forms derived from MSR waste streams) or WG 8 (Salt, Granular Solids, Powder). Table 5-5 and Table 5-6 show WG 4 and WG 8 summary evaluations, respectively, for four generic repository concepts under consideration in the U.S. Program.

It is worth mentioning that some of the engineered waste forms (e.g., glass sodalite made from salt waste), have not been demonstrated at scale. Additionally, this type of processing will generate additional (secondary) waste. These issues are represented in Table 5-5 by the yellow shaded columns for Secondary Waste Generation and Technical Readiness.

Table 5-5. Summary of Other Engineered Waste Forms, which could serve as a precedent for glass or ceramic waste forms from an MSR (from SNL, 2014)

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	✓	○	○	✓
Crystalline	✓	○	✓	○	○	✓
Clay/Shale	✓	✓	✓	○	○	✓
Deep Borehole	✓	○	✓	○	○	✓

Legend:	✓ Strong	○ Moderate	● Weak/Uncertain	✕ Not Feasible

Table 5-6. Summary of Salt, Granular Solids, and Powders (from SNL, 2014)

Disposal Concept	Disposal Option Performance	Confidence in Expected Performance Bases	Operational Feasibility	Secondary Waste Generation	Technical Readiness	Safeguards and Security
Salt	✓	✓	○	✓	✓	○
Crystalline	✓	○	○	✓	✓	○
Clay/Shale	✓	✓	○	✓	✓	○
Deep Borehole	✓	○	○	✓	○	○

Legend:	✓ Strong	○ Moderate	● Weak/Uncertain	✕ Not Feasible

The operational feasibility reflects uncertainty of handling, storing, transporting, and emplacing this waste form, while the yellow for Confidence in expected Performance Bases for Crystalline and Deep Borehole, reflect uncertainty with regards to the longevity of the waste form in the disposal environment (in disposal concepts that place higher reliance on the engineered system, including the waste form). The yellow in

Safeguards and Security reflects the risk potential for these wastes to be easily dispersed (e.g., in a terrorist attack).

Previous experience with MSR operation is limited to the Molten Salt Reactor Experiment (MSRE), which operated between June 1965 and December 1969, before it was shut down permanently. The MSRE was a graphite-moderated, liquid-fueled reactor that used a fuel formed by dissolving UF_4 fuel in a carrier salt composed of a mixture of LiF, BeF_2, and ZrF_4 (Bechtel Jacobs, 2009). The experiment began with ^{235}U as the fissile material, but ^{233}U and ^{239}Pu were also studied as fuels in the MSRE over its lifetime (Morgan & Johnson, 2019). The fuel salt was circulated by a fuel salt pump, at temperatures above 600°C, through the reactor vessel and a primary heat exchanger (Morgan & Johnson, 2019). Heat was transferred from the fuel salt to a secondary coolant salt in this primary heat exchanger (Bechtel Jacobs, 2009).

5.2.3.1 Storage and Transportation

The molten salt used in the MSRE at Oak Ridge National Laboratory (ORNL) was to be transferred to salt canisters via the existing drain line. However, the transfer was not successful, likely due to a blockage in the transfer line caused by a mixture of material that remained solid at the melting temperature of the salt (Bechtel Jacobs, 2009). As of 2019, the salt, from which uranium has been removed (Bechtel Jacobs, 2009) was still in the reactor drain tanks (Morgan & Johnson, 2019). Therefore, it is not currently in a storage system, nor has it been transported.

Radiolysis of the fluoride salt generates fluorine gas, which must be removed from the drain tanks one to two times per year to maintain the pressure below the safety basis limit (Morgan & Johnson, 2019). Before uranium was removed from the salt, fluorine gas combined with U in the fuel to form gaseous UF_6, which accumulated in off-gas filters or other locations (Peretz et al., 1998), creating concerns with respect to criticality safety, materials control and accountability, and radiation hazard (Flanagan et al., 2018). Collection of this gas in charcoal filters led to formation carbon-fluorine reaction products (C_xF), which has the potential to decompose explosively (DelCul et al., 2000).

Delineating storage and transportation issues are challenging due to the limited BENFC experience associated with molten salt reactor waste streams. In SNL (2014), discussion of such materials included potential for respirable fines during storage and or transportation activities, especially under off-normal conditions/events. Conceptual planning has been carried out that includes specifics on the waste form characteristics, as well as secondary waste streams. These plans include treatment, conversion, and/or stabilization of the MSR spent fuel and other waste streams into forms that can be readily stored and transported in expected environments. However, there are analyses that would be performed in both the storage and transportation gap analyses to ensure safety in terms of the maintenance of confinement boundaries. These analyses should include fuel confinement, thermal, criticality, and shielding considerations, particularly taking into account the characteristics of molten salt reactor spent fuels and waste streams to ensure that storage and transportation regulations are met for potential spent fuels and waste forms (NASEM, 2022).

5.2.3.2 Disposal Challenges

Disposal of spent salt fuel that has solidified and is no longer molten may be possible in a salt repository but could present challenges for other geologies because the salt waste form will dissolve easily in water, making radionuclides available for transport away from the waste package. In addition, if not treated, the fluoride salt will continue to generate fluorine gas as a result of radiolysis (Flanagan et al., 2018), pressurizing the waste package. Another source of gas includes ^{3}H if a lithium salt is used. Finally, chloride salts composed of ^{35}Cl can produce ^{36}Cl while in the reactor, and ^{36}Cl is a long-lived radionuclide that would have to be considered as part of the post-closure performance assessment.

Other waste streams associated with molten salt reactors (MSRs) and ways to manage and treat them have been identified (Riley et al., 2018). These waste streams include off-gas streams, separated salt streams, metal waste streams, carbon waste streams, decommissioning and decontaminating waste streams, and operating waste streams.

5.2.3.3 Salt-Fueled Molten Salt Reactor Fuel Cycle (Once-Through, Treated)

Among the various fuel cycles considered here, the waste streams/waste forms for the salt-fueled MSR fuel cycle are likely the least mature due to the lack of specific information regarding material/chemical/physical form and radionuclide content of the intended waste forms for each proposed waste stream (e.g., dimensions, mass, and containerization characteristics). Spent fuel salts from MSRs are usually regarded as unsuitable for direct disposal in most generic mined repository concepts due to their water solubility; the only notable exception might be for direct disposal of spent fuel salts in bedded-salt repositories, owing to the absence of flowing or circulating waters in thick salt formations (NASEM et al., 2020).

For typical salt-fueled MSRs, the spent fuel salt is expected to be directly discharged into subcritical storage tanks or decay canisters and allowed to cool and solidify before possible treatment/processing. Note that following the shutdown of the seminal fluoride salt-fueled MSRE reactor, the used fuel salt was drained into two criticality-safe storage tanks, while the flush salt was drained into a separate tank. The MSRE used fuel salt waste, along with numerous MSRE components, remain in extended storage-in-place at ORNL, awaiting final disposition (Fredrickson *et al*., 2018). Final disposal of the salt waste was not addressed in the record of decision for interim action to remove fuel and flush salts from the MSRE facility (DOE, 1998; DOE, 2006b), because of the high uncertainty or speculative nature of the final disposal locations and waste acceptance criteria for those locations (Ougouag et al., 1996). Thus, limited experience and information exists on actual waste disposition pathways and waste form characteristics for such salt-fueled MSRs.

Although salt-fueled MSRs are highly flexible and can be configured into modified open or full-recycle configuration, modified open MSR fuel cycles do not usually include chemical processing of the fuel salt and fully open MSR fuel cycle is not an option since the gaseous fission products inherently separate from the liquid fuel (Holcomb *et al*., 2011). On-line reprocessed reactor designs may also have an increased potential for small-scale radioactive material leaks and increase the plant capital costs due to the need for a hot-cell type environment for fuel salt reprocessing (Holcomb *et al*., 2011). Therefore, for the sake of simplicity, this discussion is limited to salt-fueled MSR fuel cycle once-through, treated. Additional possible MSR fuel cycle options have been proposed, for example, by Riley and co-workers (Riley *et al*., 2018).

No criticality concerns exist for these short-lived waste-form materials, which are not classified as HLW, although no specifics are currently known regarding the types and amounts of radionuclides collected in those materials. Limited or no information is available on the physicochemical and inventory characteristics of these materials (e.g., WF dimensions, packaging options, FP concentration and distribution in these WF materials). Such scarcity of information limits the current ability to assess whether and how those WF can be disposed of in generic mined repository concepts or in deep boreholes; the latter option is highly dependent upon WF dimensions and packaging size, with additional treatment needed if dimensions exceed typical deep borehole diameters. If additional treatment of molten salt spent fuel is utilized for disposal purposes, a fifth waste form can be envisioned from the waste stream consisting of mixtures of metal chloride or fluoride salts containing unknown amounts of radionuclides (see Table 3-12, WF Classification column, 5a, 5b, and 5c)).

Depending on the options/concepts used for treating molten salt spent fuel, three main types of HLW WF may result from treatment: phosphate glass (WF 5a. Option 1); Ceramic-metal composite (WF 5b. Option

2); glass-bonded ceramic (WF 5c. Option 3). In comparison to WF 1 through 4 discussed above, it can be inferred that WF 5a-c will have relatively longer lifetimes in the generic disposal environments under consideration, although knowledge gaps remain and WF performance should be further investigated. For example, more constraints are needed to assess the degradation of metallic components of WF 5b under oxidizing unsaturated conditions and understand the impact of local redox conditions on WF degradation in the various mined repository concepts. The lack of information on the radionuclides inventory and fissile loading from the MSR spent nuclear fuel significantly complicates the assessment of WF 5a-c for the disposal options under consideration, in particular medium to high issues related to WF thermal output and criticality and their impact on the various disposal concepts. Finally, the current scarcity of information regarding the dimensions and package sizing/containerization of those WF limits the assessment of their eligibility for deep borehole disposal.

Several evaluation criteria and associated metrics are listed below for disposal of salt waste forms relevant to the salt-fueled MSR fuel cycle:

Disposal Option Performance:

(1). *Likely to comply with long-term standards?*: Probably for deep borehole disposal, maybe for disposal in a salt repository, and probably not for other disposal environments because this waste form would have no appreciable lifetime once the waste package failed. The question then becomes whether the waste form(s) produced by treatment of the salt are likely to comply with long-term standards. Thermal, chemical, physical, and packaging characteristics would come into play.

(2). *Identification of key attributes of the disposal system:* This waste form has the potential to have a very high thermal load per volume, particularly the salt containing the fission products that are removed through online reprocessing; therefore, the thermal response of the disposal system (deep borehole or (maybe) salt) would be important. The thermal load of the waste forms produced by treating would be a key consideration, as would their chemical and physical characteristics.

Confidence in Expected Performance Bases:

(1). *Additional EBS components needed above baseline for each design concept*: None (if disposed of directly); otherwise, this would depend on the waste forms produced by treatment of this fuel.

(2). *Robustness of information bases; simplicity vs. complexity; knowledge gaps*: This waste form has not been studied in terms of its long-term performance. Waste forms created from reprocessing have the potential to be engineered and well characterized. The information base for borehole disposal is not as substantial as that for other disposal options.

Operational Feasibility:

(1). *Ease in ensuring worker health and safety at all stages*: This waste form may have respirable fines that would have to be managed; however, this would be of concern only in a low-probability accident event. For disposal in anything but a deep borehole or (maybe) a salt repository, this fuel will have to be reprocessed, making it more difficult to avoid potential increases in worker dose. The composition of the final waste forms would be a factor.

(2). *Special physical considerations at any stages based on physical characteristics*: May need special equipment for transporting respirable fines. Salt waste is also known to generate gas as a result of radiolysis (see Section 5.2.3), which would have to be managed.

Secondary Waste Generation:

(1). *Amount of LLW generated during handling and treatment*: Reprocessing this waste would probably generate LLW as well as non-standard waste forms that need classification (i.e., LLW, HLW, SNF).

(2). *Amount of mixed waste generated*: Analysis of proposed waste form(s) needed. If the waste treatment process creates a uranium product, management of this product would be a consideration.

Technical Readiness:

(1). *Status of needed waste form generation technology*: The processes that might be used to create a more stable waste form have not yet been defined.

(2). *Status of transportation and handling systems*: A waste form of this type is currently being stored in ORNL's reactor dump tanks; it has been a challenge to remove it from these tanks. At INL, the salt waste from sodium-bonded spent fuel (which is similar to this waste) is being "stored" in the unit that is processing the sodium-bonded spent fuel and is stuck there because it has no storage option.

(3). *Status of disposal technologies*: Salt disposal concept is ready. Deep borehole disposal concept is not ready. Other disposal concepts are ready but are not compatible with this waste; it would have to be treated to be disposable in the other types of repositories.

Safeguards and Security:

(1). *National security implementation difficulty (fissile content)*: Further study of potential waste forms needed. High fissile content waste forms would likely have increased MC&A and safeguards.

(2). *Radiological dispersion device prevention implementation difficulty*: Further study of this waste form and waste forms produced by reprocessing this waste is needed with respect to creating a radiological dispersion device. Is the salt easily dispersible or is it melted/crystallized together to form a hard mass?

5.3 Higher Enriched LEU

Many of the ARs and AFs are proposing to use uranium-based fuel with enrichments between 5 percent and 19.75 percent (HALEU). There is considerable overlap between the disposition challenges of HALEU and ATF because ATF typically include fuels that have increased enrichment and/or higher burnup (see NRC 2022b). The disposition challenges discussed below also generally apply to ATF (see Section 5.4).

5.3.1 Storage and Transportation

The challenges introduced by HALEU are evaluated for the Storage, Transportation, and Disposal challenges (i.e., those outlined in Ch. 4 for Storage and Transportation). This includes evaluating the impact of increased fuel enrichment, which may result in increased thermal output and higher burnup, and as listed in Ch. 4:

- Thermal profiles

- Stress Profiles
- Drying Issues
- Fuel Transfer
- Criticality Safety Analyses

A full description of high, medium, and low priority Storage and Transportation Challenges can be in found in Ch. 4

The higher enrichment of the fuel might entail different packaging designs to maintain subcritical conditions for handling, storing, and transporting the spent fuel because of its higher reactivity. There is experience in this area, as the DOE has transported and is currently storing many different spent fuels with enrichments between 5 percent and 19.75 percent, as well as spent fuels that have much higher enrichments (DOE, 2010).

5.3.2 Disposal

As with Storage and Transportation, the increased enrichment and thermal output and higher burnup od HALEU could affect the FEP analyses as described in Ch. 4 and will be considered as part of the detailed evaluations. This includes changes to the source term, which may be affected by:

- Corrosion processes (both fuel and packaging)
- Package physical form and early failure
- Radionuclide mass fractions
- Waste form degradation processes
- Chemical interactions with groundwater (both fuel and package)
- Radionuclide solubility and sorption
- Colloid formation (heavily influenced by fuel and package materials and degradation)
- Radioactive decay and ingrowth (which would vary by fuel type, burnup, etc)
- Criticality Analyses
- Pyrophoricity

Assessment of disposal issues for HALEU will include investigation into effects on the post-closure source-term and criticality considerations over relevant repository timescales (e.g., 10,000 and up to 1,000,000 years). Note that several highly enriched DOE-managed SNF types (up to 93.5 wt.% U-235) were included as minor components of the inventory in the disposal plan for Yucca Mountain (DOE, 2010). A full description of Disposal Challenges can be in found in Ch. 4

5.4 Accident Tolerant Fuels

ATFs refers to new fuel pellet and cladding designs that can potentially increase plant performance and improve accident conditions. New fuel pellet and cladding designs and materials have been proposed and are being studied for use in next generation nuclear plants of the current fuel cycle. A detailed study of the effects of ATF on the BENFC (disposal) would be conducted to define the specific gaps regarding the ATFs. Below is a brief discussion on proposed new pellets and cladding materials, and preliminary assessment of the potential effects to disposal.

5.4.1 Fuels

The following are examples of fuel types that have been proposed as ATFs:

- Doped fuels have been proposed to increase performance, such as . Among the reported benefits are improved mechanical properties and corrosion resistance. These dopants may also reduce fuel pellet swelling/fission gas release by increased grain size. Such improvements would also benefit disposal in terms of performance of the waste form and the waste package.
- Use of higher-enriched fuels has also been proposed to increase burnup. For disposal this needs to be studied for the effect of thermal limits and loading, effect of higher dose, effect on waste package failure mechanisms, effect on waste form degradation.
- Use of metallic fuels to increase thermal performance in-reactor due to higher thermal conductivity. For disposal use of metallic fuels needs to be studied for the effect of thermal-loading, effect of waste form corrosion, effect of swelling and resulting effects on waste package integrity

5.4.2 Cladding

Use of different cladding materials (such as those referenced in Section 3.2.1.2) are reported to reduce corrosion (mitigate oxidation of zirconium), which may also benefit disposal by providing a more durable waste-form.

Carmack et al. (2013) mentions the BENFC with ATF and that ATF must maintain the storage and repository performance of the fuel; if not, the storage and disposal engineering solutions must be augmented (assuming a once-through fuel cycle).

Saltzstein et al., (2021) pose many questions regarinding the ATF gaps for the back end of the nuclear fuel cycle.

- Hotter decay fuel effect on the mechanical integrity of the fuel, cladding, and other structures, systems, and components over long-term storage and transportation
- The structural integrity of the ATF during storage and transportation
- Effectiveness of water removal process compared to the traditional fuel vs ATF for drying
- Study of criticality potential with ATF in the current cask/canister design
- Limitations on current equipment to handle potentially heavier and more brittle ATF loads

Additional topics on the behavior of ATF to further evaluate gaps for storage, transportation, and long-term disposal include:

- Form
- Size distribution
- Radioactivity
- Fuel fragmentation
- Cracking
- Oxidation
- Cladding strength and ductility
- Creep
- Embrittlement

Bragg-Sitton discusses different types of ATF that are being developed in the U.S. (Bragg-Sitton, 2014) and indicates that

1. Los Alamos National Laboratory (LANL) is developing corrosion-resistant metallic alloys such as iron-chromium-aluminum (FeCrAl) and molybdenum (Mo) as cladding for high-density ceramic fuels.

2. ORNL efforts in oxidation-resistant steel cladding and TRISO-based fuel.

3. Pacific Northwest National Laboratory (PNNL) is studying U-Mo Fuel concept for LWR.

4. Areva has developed ATF concepts to reduce fission gas generation by chromia doping, improve safety margin and increase heat transfer by applying SiC fibers, and coatings on the Zr-alloy cladding to increase coping time during accident conditions.

5. General Electric is working with LANL to replace Zr-alloy cladding with advanced steel.

6. Westinghouse is looking at improving the economics of ATF fuels, and

7. cladding studies of ATF are led by the University of Illinois and the University of Tennessee.

Additional research gaps to be filled for the backend of the fuel cycle needs incldue the lack of data for potential storage and transportation evaluations for ATF from the expected higher burnups of ATF, which translate to higher temperatures and higher internal rod pressures. . The degradation characteristics of ATF are yet to be defined, such as clad-coating robustness, potential corrosion, and hydride embrittlement in potential in areas of damaged clad coatings. The TRISO fuel design (pebbles) that ORNL is considering may lead to new designs for storage, transportation, and disposal containers, or at least differing baskets for holding/cushioning/supporting the TRISO fuel.

Geelhood and Luscher (2019) have collected data available on ATF on chromium coated zirconium alloy cladding concept. In section 6.2 of the paper, they discuss ex-reactor data collected on unirradiated samples. The thermal properties of that particular ATF are undefined; however, there are summaries of data on ballooning, high-temperature corrosion, fretting, ductility, and autoclave corrosion testing.

Overall, given that ATFs are similar to the traditional Zr-alloy fuels (except for the TRISO fuel design), the strategy for assessing ATF can be viewed similarly to that for LWR fuel for storage, transportation, and long-term disposal. The gaps for assessing ATF are fewer are more readily delineated compared to AR SNF.

5.4.3 Uranium alloy fuels

*Note: For general metallic fuel disposition challenges, see Section 5.2.2.

5.4.3.1 Storage and Transportation

The reaction of residual water with uranium alloy fuels is much more aggressive than with the uranium oxide-based fuels. Metallic fuels exhibit significant swelling (Honnold et al., 2021), which might affect cladding performance during storage and transportation.

5.4.3.2 Disposal

These fuels are currently being studied. These fuels were part of the disposal plan for the proposed Yucca Mountain repository (DOE, 2010). Uranium alloy fuels that contain metallic sodium would undergo treatment prior to disposal. Metallic fuels exhibit significant swelling (Honnold et al., 2021), which might affect cladding performance during the post-closure period.

5.4.4 Uranium nitride fuels

5.4.4.1 Storage and Transportation

The reaction of residual water with uranium nitride fuels is much more aggressive than with the uranium oxide-based fuels. The ATF/AF gap analysis (Honnold et al., 2021) also pointed out that nitrides are more reactive than oxides, which will be assessed in FEP analyses for generic disposal environments. These fuels have a higher thermal conductivity and different physical characteristics (e.g., mass); they tend to be higher burnup fuels, leading to increased pellet swelling, higher temperatures, finer particles, and higher doses, with and without canister failure; release of volatile fission products from fuel pellets may be a concern; and UN-based fuel displays poor oxidation resistance. These issues may also apply to storage environments, although to a lesser degree due to drying of spent fuel prior to placement in a dry canister storage system. However, the minimum water content threshold that would be sufficient for safe storage of uranium nitride fuels has not yet been determined.

5.4.4.2 Disposal

Plutonium-uranium nitride spent fuel was part of the disposal plan for Yucca Mountain (DOE, 2010), so it was at least considered for disposal. Generation of minor actinides in the nitride fuel production process is not fully explored in large-scale applications (Honnold et al., 2021). Production of ^{14}C when ^{14}N absorbs a neutron and emits a proton (unless ^{15}N is used) may be a concern because of the mobility of ^{14}C in a repository.

5.4.5 Uranium silicide fuels

5.4.5.1 Storage and Transportation

These fuels are currently being studied. The gap analysis (Honnold et al., 2021) identified higher thermal conductivity as a difference between this fuel and traditional UO_2 fuels. If this fuel is burned more than traditional UO_2 fuels, this could lead to increased pellet swelling higher temperatures, finer particles, and higher doses, with and without canister failure. It could also lead to release of volatile fission products from the fuel pellet.

5.4.5.2 Disposal

Fuels of this type were part of the disposal plan for the proposed Yucca Mountain repository (DOE, 2010). No disposal challenges or gaps have been identified as of yet.

6 CONCLUSIONS

This report has reviewed and analyzed BENFC technical gaps that may exist for deployment of Advanced Nuclear Reactor concepts. In addition to storage, transportation, and disposal analyses, the relevant regulatory considerations were reviewed. The following sections will summarize Chapters 2 through 5 of this report. The final two sections provide an overall summary and identification of the major technical gaps identified by the analyses. A summary of key findings is provided first.

6.1 Key Findings

- A review of Regulatory Considerations was conducted. Given the differences of AR SNF and AR waste streams from the characteristics of Light Water Reactor (LWR) SNF, updates or supplements to methodology/guidance may facilitate application of existing regulations to evaluations for ARF and ARWS.

- The existing *regulatory guidance* is largely based on the low-enriched zircaloy clad uranium-oxide SNF. Updates to guidance may facilitate the application of existing regulations to address BENFC assessments for ARF and ARWS.

- Uncertainty around the specifics of AR designs can be the result of specifications being considered proprietary information or because the concept is still at a low level of technical readiness. This uncertainty presents difficulties to conduct full and thorough of analysis of ARF and ARWS disposition.

- Based on a survey of reactor types, three fuel types are most common: TRISO, Sodium-bonded Metallic, and Salt Fuel. Of these three, literature review and analysis indicate that TRISO is the only AR Fuel with a relatively clear pathway for direct disposal, while salt fuel waste is possible in certain repository design concepts. Sodium-bonded metallic fuels must be processed and/or treated, as the metallic sodium bond reacts violently with water, which presents hazards not only for disposal, but can also present potential issues for storage and transportation.

- While TRISO fuel has a clear direct disposition pathway, it should be noted that previous BENFC experience with TRISO fuel is related to prismatic TRISO from the Ft. St. Vrain reactor, and some emerging commercial concepts employ TRSIO pebbles in a Pebble Bed Reactor (PBR). Fuel compacts removed from the prismatic block may necessitate further analysis, especially with respect to safeguards and security and packaging and handling. TRISO from a molten salt-cooled PBR may also experience an ingress of salt into the graphite/coated particle matrix, so additional investigation will be performed in this area.

- Many commercial designs plan to use High Assay Low-Enriched Uranium (HALEU) fuel, which has a higher burn-up than traditional light-water reactor (LWR) fuels. This has implications across the BENFC in storage, transportation, and disposal, as higher thermal loads tend to accelerate degradation mechanisms. There are also notable implications for critical safety associated with HALEU.

- Accident tolerant fuels (ATFs) are generally expected to be more robust in the disposal environment. However, thermal and mechanical properties of ATFs are sufficiently different such that further analyses are warranted, especially for impacts on storage and transportation. The additional elements added to ATF have the potential to impact near-field geochemistry in the disposal setting. Because there are a variety of ATFs, scoping studies should be done on a case-by-case basis as necessary, for impacts across the BENFC.

6.2 Summary of Regulatory Considerations

The existing *regulations* for storage and transportation of SNF and HLW are generally agnostic to specific fuel or waste characteristics (with the exception of the information needed for classification). Therefore, updates or supplements to the existing regulation are not anticipated for ARF and ARWS.

The existing *regulatory guidance* is largely based on the low-enriched zircaloy clad uranium-oxide SNF. Significant updates or new guidance are suggested to explicitly address BENFC needs for ARF and ARWS. The primary technical concepts for consideration in updating regulatory guidance relate to the following areas:

- Storage confinement
- Transportation containment
- Retrievability during storage
- Geometry control during transportation
- Thermal and structural performance criteria during loading, storage, and transportation operations
- Burnup credit and criticality control
- Source term development

The guidance is expected to be highly dependent on the diversity of fuel designs and reactor irradiation characteristics. Therefore, some of the guidance (e.g., burnup credit) would likely not be available until significant reactor irradiation experience has been gained.

6.3 Summary of Waste Characteristics Important to BENFC

Previous analyses of waste characteristics important to waste disposition were revisited and analyzed in the context of ARF and ARWS. The following characteristics were analyzed:

- Radionuclide inventory
- Thermal Output
- Chemical Characteristics
- Physical Characteristic
- Packaging and Criticality

6.4 Technical Gaps related to Advanced Reactor Secondary Waste Streams

In many cases, analysis of secondary Advanced Reactor Waste Streams is difficult due to the state of available information on design specifics and performance data from irradiated fuel. The identified gaps can be summarized as follows:

6.4.1 General Gaps

- Secondary products at fuel-WF and WF-EBS interfaces in the context of AR fuels are not known.
- Gas solubility data in complex multi-component salt waste envisioned to be produced by MSRs (needed for performance and safety assessment of salt waste in various repositories).

- Protection of proprietary information on innovative AFs under development by various advanced nuclear reactor and fuel companies contributes to the lack of clear disposal strategies.
- Uncertainty regarding the operating conditions, refueling needs, and performance of some of AR concepts, which can significantly affect the type and quantity of waste generated.

6.4.2 TRISO Fuel Gaps

- Waste associated with TRISO fabrication.
- Issues related to graphite superstructure of spent TRISO fuel.
- TRISO fuel used in an MSR may be affected by salt ingress into pebbles/pellets from exposure to salt coolant. Further investigations into effect of this on degradation are needed.

6.4.3 Sodium-bonded Fuel Gaps

- Waste associated with fabrication of sodium-cooled fast reactor metal fuel.
- Pyroprocessing of reactor off-gases from sodium-cooled fast reactor SNF.
- End-of-life waste stream from pyroprocessing of sodium-cooled fast reactor SNF.
- Sodium-bonded SNF from sodium-cooled fast reactors SNF (once-through, treated).
- Disposal of filters from off-gas from melt-dilute treatment of sodium-bonded SNF from sodium-cooled fast reactors SNF (once-through, treated).

6.4.4 Molten Salt Reactor Gaps

- SNF streams from salt-fueled MSRs (once-through, treated).
- Off-gases from SNF treatment from salt-fueled molten salt reactors (once-through, treated).
- Uncertainty of proposed/potential AR fuel waste forms: characterization, degradation behavior, chemical reactivity, and lifetime.

6.5 Summary Gaps related to the Disposition of Advanced Reactor Fuels

TRISO, Metallic Fuels, and Molten Salt Fuels are the fuels most likely used by ARs, based on the survey of commercial reactor AR concepts presented in Section 3.3. These fuels, HALEU, and ATFs were analyzed for Storage, Transportation, and Disposal Gaps. Precedent disposal pathways were identified based upon similar fuels and/or waste forms from those existing in the DOE inventory that have been previously analyzed. The identified gaps are summarized for each fuel type below.

6.5.1 TRISO

6.5.1.1 Storage

- Aging management concerns associated with the silicon carbide during extended storage.
- Storage of TRISO-based fuels may need to account for salt contamination after removal from salt-cooled reactors.

6.5.1.2 Transportation

- Higher fuel enrichments from the use of HALEU in TRISO fuels necessitate criticality analyses to ensure regulatory compliance.
- Higher burnups expected from reactors using TRISO fuels require thermal analyses to ensure that the 50 °C 10 CFR 71 transportation cask surface temperature limit is not reached.
- Transport of TRISO-based fuels may need to account for salt contamination after removal from salt-cooled reactors.

6.5.1.3 Disposal

- Measures may need to be taken to ensure disposal criticality control.
- No experience with actual disposal of spent coated fuel particles.
- Limited information for deep borehole disposal; analysis of graphite blocks as secondary waste needed
- Generation of ^{14}C and ^{3}H from radiolysis of graphite may need consideration.
- Gas generation from carbide exposure to water.
- Disposal of TRISO-based fuels may need to account for salt contamination after removal from salt-cooled reactors.

6.5.2 Metallic Fuel

6.5.2.1 Storage

Thermal, mechanical, corrosion, embrittlement characteristics of metallurgical bonding between fuel and cladding have not been fully evaluated

6.5.2.2 Transportation

Feasibility of transport of multi-canister overpacks storing metallic fuel (e.g., N-reactor fuel) under 10 CFR 71 has not been fully evaluated

6.5.2.3 Disposal

- Build-up of gas and swelling of the fuel may need to be assessed for post-closure performance.
- Operational feasibility and technical readiness represent major issues for most disposal concepts

6.5.3 Sodium-bonded Metallic Fuel

6.5.3.1 Storage

Effects of residual water on metallic sodium in storage canisters have not been fully evaluated (only one case study currently available for analysis); details on storage canisters currently used for sodium-bonded metallic fuels not publicly available; parameters relevant to NRC licensing of storage facilities for sodium-bonded metallic fuel have not been fully evaluated.

6.5.3.2 Transportation

Parameters relevant to NRC licensing of transportation packages for sodium-bonded metallic fuel have not been fully evaluated.

6.5.3.3 Disposal

Limited experience with treatment of salt waste produced by the EMT to produce a ceramic waste form; salts might be suitable for disposal in a salt repository but present challenges for disposal in non-salt formations; research is needed regarding whether waste forms produced by treatment of the sodium-bonded spent fuel comply with long-term disposal standards.

6.5.4 Molten Salt Fuel

6.5.4.1 Storage

Technical feasibility of storage of molten salt has not been fully evaluated (no molten salt currently being stored); extent of UF_6 generation issues associated with criticality safety, materials control and accountability, and radiation hazards have yet to be fully quantified.

6.5.4.2 Transportation

Technical feasibility of transportation of molten salt has not been fully evaluated (no successful demonstration of transportation of molten salt has been realized); extent of UF_6 generation issues associated with criticality safety, materials control and accountability, and radiation hazards have yet to be fully quantified.

6.5.4.3 Disposal

Disposal of spent salt fuel that has solidified and is no longer molten may be possible in a salt repository but could present challenges for other geologies; if not treated, fluoride salt will continue to generate fluorine gas; another source of gas includes ^{3}H in lithium salt; ^{35}Cl-containing salt can produce long-lived radionuclide ^{36}Cl that need to be assessed for post-closure performance; secondary waste generation, technical readiness, and safeguards and security may be major issues for all disposal concepts.

6.5.5 HALEU

6.5.5.1 Storage

Gaps associated with fuel form rather than HALEU, but higher enrichment may entail new storage system designs to maintain subcriticality due to higher reactivity. Higher thermal output and higher burnup may impact thermal profiles, and stress profiles, as well as present issues with respect to drying and fuel transfer and handling.

6.5.5.2 Transportation

Gaps associated with fuel form rather than HALEU, but higher enrichment may entail new transportation system designs to maintain subcriticality due to higher reactivity. Higher thermal output and higher burnup may impact thermal profiles, and stress profiles, as well as present issues with respect to drying and fuel transfer and handling.

6.5.5.3 Disposal

Disposal of HALEU could necessitate investigation into post-closure source term and criticality impacts that may occur over repository timescales

6.5.6 Accident Tolerant Fuels

6.5.6.1 Storage

- Hotter decay fuel effect on the mechanical integrity of the fuel, cladding, and other structures, systems, and components over long-term storage.
- Structural integrity of the ATF during storage; effectiveness of water removal process compared to the traditional fuel vs ATF for drying.
- Study of criticality potential with ATF in the current storage cask/canister designs; limitations on current equipment to handle potentially heavier and more brittle ATF loads.
- More research needed to fully evaluate ATF fuel and cladding characteristics, such as form, size distribution, radioactivity, fuel fragmentation, cracking, oxidation, cladding strength and ductility, creep, and embrittlement.

6.5.6.2 Transportation

- Hotter decay fuel effect on the mechanical integrity of the fuel, cladding, and other structures, systems, and components during transportation.
- Structural integrity of the ATF during transportation; study of criticality potential with ATF in the current transportation cask/canister designs.
- Limitations on current equipment to handle potentially heavier and more brittle ATF loads; more research needed to fully evaluate ATF fuel and cladding characteristics, such as form, size distribution, radioactivity, fuel fragmentation, cracking, oxidation, cladding strength and ductility, creep, and embrittlement.

6.5.6.3 Disposal

- ATF/AF fuels with higher density and enrichment may have increased criticality risk compared to traditional LWR fuels and may need criticality control
- A detailed study of the disposal of ATF would need to be conducted since limited information exists
- Uranium alloy fuels: Uranium alloy fuels that contain metallic sodium would undergo treatment prior to disposal
- Uranium nitride fuels: Production of ^{14}C may be a concern because of the mobility of ^{14}C in a repository
- Uranium silicide fuels: No disposal challenges or gaps have been identified

A summary of AR fuels, by fuels type, is presented in Table 6-1. Storage, Transportation, and Disposal Gaps are summarized in Table 6-2.

Table 6-1. Summary of Advanced Reactor Fuels Characteristics

Spent Fuel	Prior or Current Experience in BENFC	Properties of Concern	Additional Wastes Associated with Fuel Cycle (beyond that associated with a typical LWR)	Additional Radioisotopes or Chemical Constituents of Concern
TRISO (PMR)	Has been transported; is currently stored under NRC license (Fort St. Vrain; NWTRB, 2020). Was part of the disposal plan for Yucca Mountain (DOE, 2010).	High enrichment – post-closure criticality control needs to be considered.	Used graphite prismatic blocks (for prismatic block type designs).	^{14}C and ^{3}H from irradiating graphite and lithium, which is an impurity in graphite. (Kitcher, 2020a)
TRISO (PBR)	US inventory lacks experience with this spent fuel type, aside from TRISO (PMR), some of which may be relevant. Germany and China have operated HTGRs	For Molten Salt PBR using TRISO, diffusion of molten salt coolant into TRISO pebble has not been studied for BENFC impacts	For Molten Salt PBR using TRISO, cooling salt waste and off gas streams	See Molten Salt below Graphite concerns for TRISO (PMR)
Metallic fuel	Several forms of metallic fuel (N reactor, Fermi fuel, etc.) have been produced, processed, stored, and transported in the U.S. These metallic fuels were included in the disposal plan for Yucca Mountain.	Potential for increased degradation and, hence source term, relative to LWR fuel.	None identified. If processing and/or treatment were deemed necessary, analysis would be needed.	Possibly upon treatment and/or processing (if necessary).
Sodium-bonded metallic fuel	Has been transported; is currently stored at INL; is being treated via EMT (NWTRB, 2017)	Contains pyrophoric metallic sodium; undergoes treatment for disposal	Would depend on treatment process (Kitcher, 2020b, DOE, 2000)	Would depend on treatment process (Kitcher, 2020b)
Molten salt (also maybe molten salt as a coolant but not a fuel)	Has not been transported or "stored;" is still in the molten salt reactor drain tanks at ORNL (Bechtel Jacobs, 2009)	Waste form dissolves easily in water. Presence of ^{233}U if thorium-based system For molten salt that is a coolant, could be contaminated with fission products from defective	Off-gas waste streams	Volatile fission products and their decay daughter nuclides, particulates, tritium, water, oxides, nitrides, halides, trace amounts of salt in the form of aerosols, and noble gases (Kitcher, 2020c) ^{3}H if a lithium salt is used Formation of ^{36}Cl

Spent Fuel	Prior or Current Experience in BENFC	Properties of Concern	Additional Wastes Associated with Fuel Cycle (beyond that associated with a typical LWR)	Additional Radioisotopes or Chemical Constituents of Concern
		fuel particles (Andrews et al., 2021)		Formation of UF^6 (Flanagan et al., 2018)
Higher enriched LEU	Applies to multiple fuel forms. In general, was part of the disposal plan for Yucca Mountain (DOE, 2010).	High enrichment – criticality during storage, transport, and disposal, source term during disposal Higher burnup, cycle length, and power – fuel integrity during Storage and transportation impact on disposal source term esp. degradation mechanisms and radionuclide mass fractions and radionuclide transport/sorption mechanisms	None identified	None identified
Cr-doped LEU UO_2 pellets	Currently being studied	Reduced fission gas release, improved performance with respect to both corrosion resistance, and pellet-cladding interaction (Arborelius et al., 2006)	None identified	More Cr in spent fuel than in the past

Spent Fuel	Prior or Current Experience in BENFC	Properties of Concern	Additional Wastes Associated with Fuel Cycle (beyond that associated with a typical LWR)	Additional Radioisotopes or Chemical Constituents of Concern
Uranium alloy fuels (such as Mo and Zr)	Currently being studied. Was part of the disposal plan for Yucca Mountain (DOE, 2010).	May contain metallic sodium, would undergo treatment for disposal Metallic fuels exhibit significant swelling (Honnold et al., 2021) - how would that affect cladding? May also have increased enrichment compared to typical UO_2 fuels (Honnold et al., 2021) - leads to criticality concerns for the post-closure period.	Would depend on whether there was a treatment process and what it was.	Would depend on whether there was a treatment process and what it was
Uranium nitride fuels	Currently being studied. Plutonium-uranium nitride was part of the disposal plan for Yucca Mountain (DOE, 2010).	Nitrides more reactive than oxides (Honnold et al., 2021), issue for post-closure criticality. Higher thermal conductivity (Honnold et al., 2021) Removal of residual water during drying process. Different physical characteristics (e.g., increased mass) Higher burnup fuel, leading to increased pellet swelling, higher temperatures, finer particles, and higher doses, with and without canister failure	Generation of minor actinides in the nitride fuel production process it not fully explored in large-scale applications (Honnold et al., 2021)	^{14}C produced when ^{14}N absorbs a neutron and emits a proton (unless ^{15}N is used)

Spent Fuel	Prior or Current Experience in BENFC	Properties of Concern	Additional Wastes Associated with Fuel Cycle (beyond that associated with a typical LWR)	Additional Radioisotopes or Chemical Constituents of Concern
		Release of volatile fission products from fuel pellet UN-based fuel displays poor oxidation resistance (Honnold et al., 2021)		
Uranium silicide fuels	Currently being studied Was part of the disposal plan for Yucca Mountain (DOE, 2010).	Different physical characteristics (e.g., increased mass) Higher burnup fuel, leading to increased pellet swelling, higher temperatures, finer particles, and higher doses, with and without canister failure Release of volatile fission products from fuel pellet Higher thermal conductivity (Honnold et al., 2021)	None Identified	What happens when silica absorbs a neutron?
Cr-coated cladding	Currently being studied	Load capacities and ductile/brittle behaviors. Cladding creep behavior will be different and needs to be understood (Honnold et al., 2021) Properties of irradiated cladding (e.g., hardness, ductility) Effect of residual water on cladding with cracks, scratches, or fretting Physical characteristics (e.g., increased mass)	None identified	None identified

Spent Fuel	Prior or Current Experience in BENFC	Properties of Concern	Additional Wastes Associated with Fuel Cycle (beyond that associated with a typical LWR)	Additional Radioisotopes or Chemical Constituents of Concern
		Higher burnup fuel, leading to increased pellet swelling, higher temperatures, finer particles, and higher doses, with and without canister failure		
FeCrAl cladding	Currently being studied	Load capacities and ductile/brittle behaviors Lower creep rate; may reduce pellet-cladding mechanical interaction failures (Honnold et al., 2021) Properties of irradiated cladding (e.g., hardness, ductility) Physical characteristics (e.g., increased mass) Higher burnup fuel, leading to increased pellet swelling, higher temperatures, finer particles, and higher doses, with and without canister failure Higher enrichment may be needed (Honnold et al., 2021); may lead to post-closure criticality concerns	None identified	None identified
SiC-based cladding	Currently being studied	Load capacities and ductile/brittle behaviors Cladding creep behavior will be different and needs	None identified	None identified

Spent Fuel	Prior or Current Experience in BENFC	Properties of Concern	Additional Wastes Associated with Fuel Cycle (beyond that associated with a typical LWR)	Additional Radioisotopes or Chemical Constituents of Concern
		to be understood (Honnold et al., 2021) Physical characteristics (e.g., increased mass) Higher burnup fuel, leading to increased pellet swelling, higher temperatures, finer particles, and higher doses, with and without canister failure Thermal conductivity degrades with neutron irradiation; heat transfer rates decrease over time.		

Table 6-2. Summary of storage, transportation, and disposal gaps for each advanced reactor fuel type.

Spent Fuel	Storage Gaps	Transportation Gaps	Disposal Gaps
TRISO	SiC layer failure from internal gas pressure, irradiation-induced cracking and debonding of pyrocarbon layers, fuel kernel migration, chemical attack of the SiC layer by fission products, thermal decomposition of the SiC layer, and enhanced SiC permeability and/or SiC degradation have all been identified as potential failure mechanisms, although these mechanisms are not expected to occur under typical storage conditions (Hall *et al*., 2019c).	Transportation of TRISO from salt-cooled reactors may undergo additional cleaning	Measures may need to be taken to ensure disposal criticality control; no experience with actual disposal of spent coated fuel particles; limited information for deep borehole disposal; analysis of graphite blocks as secondary waste needed
Metallic fuel	Thermal, mechanical, corrosion, embrittlement characteristics of metallurgical bonding between fuel and cladding have not been fully evaluated	Feasibility of transport of multi-canister overpacks storing metallic fuel (e.g., N-reactor fuel) under 10 CFR 71 has not been fully evaluated	Build-up of gas and swelling of the fuel may need to be assessed for post-closure performance; operational feasibility and technical readiness represent major issues for most disposal concepts

Spent Fuel	Storage Gaps	Transportation Gaps	Disposal Gaps
Sodium-bonded metallic fuel	Effects of residual water on metallic sodium in storage canisters have not been fully evaluated (only one case study currently available for analysis); details on storage canisters currently used for sodium-bonded metallic fuels not publicly available; parameters relevant to NRC licensing of storage facilities for sodium-bonded metallic fuel have not been fully evaluated	Parameters relevant to NRC licensing of transportation packages for sodium-bonded metallic fuel have not been fully evaluated	Limited experience with treatment of salt waste produced by the EMT to produce a ceramic waste form; salts might be suitable for disposal in a salt repository but present challenges for disposal in non-salt formations; research is needed regarding whether waste forms produced by treatment of the sodium-bonded spent fuel comply with long-term disposal standards
Molten salt (also maybe molten salt as a coolant but not a fuel)	Technical feasibility of storage of molten salt has not been fully evaluated (no molten salt currently being stored); extent of UF_6 generation issues associated with criticality safety, materials control and accountability, and radiation hazards have yet to be fully quantified, but HF generation is a concern.	Technical feasibility of transportation of molten salt has not been fully evaluated (no successful demonstration of transportation of molten salt has been realized); extent of UF_6 generation issues associated with criticality safety, materials control and accountability, and radiation hazards have yet to be fully quantified, but HF generation is a concern	Disposal of spent salt fuel that has solidified and is no longer molten may be possible in a salt repository but could present challenges for other geologies; if not treated, fluoride salt will continue to generate fluorine gas; another source of gas includes ^{3}H in lithium salt; ^{35}Cl-containing salt can produce long-lived radionuclide ^{36}Cl that need to be assessed for post-closure performance; secondary waste generation, technical readiness, and safeguards and security may be major issues for all disposal concepts
Higher enriched LEU	Gaps associated with fuel form rather than HALEU, but higher enrichment may require new storage system designs to maintain subcriticality due to higher reactivity; safeguards and security systems for storage being evaluated	Gaps associated with fuel form rather than HALEU, but higher enrichment may entail new transportation system designs to maintain subcriticality due to higher reactivity; safeguards and security systems for transportation being evaluated	Disposal of HALEU could increase focus on post-closure criticality and how to manage it over repository timescales; safeguards and security systems for disposal being evaluated
Accident Tolerant Fuels and Cladding	Hotter decay fuel effect on the mechanical integrity of the fuel, cladding, and other structures, systems, and components over long-term storage; structural integrity of the ATF during storage; effectiveness of water removal process during drying operations compared to the traditional fuel vs ATF for drying; study of criticality potential with ATF in the current storage cask/canister designs; limitations on current equipment to	Hotter decay fuel effect on the mechanical integrity of the fuel, cladding, and other structures, systems, and components during transportation; structural integrity of the ATF during transportation; study of criticality potential with ATF in the current transportation cask/canister designs; limitations on current equipment to handle potentially heavier and more brittle ATF loads; more research needed to fully	ATF/AF fuels with higher density and enrichment may have increased criticality risk compared to traditional LWR fuels and may have added criticality control; ; uranium alloy fuels that contain metallic sodium would undergo treatment prior to disposal; uranium nitride fuels: production of ^{14}C may be a concern because of the potential mobility of ^{14}C in an unsaturated repository; Uranium nitride fuels: reaction

Spent Fuel	Storage Gaps	Transportation Gaps	Disposal Gaps
	handle potentially heavier and more brittle ATF loads; more research needed to fully evaluate ATF fuel and cladding characteristics, such as form, size distribution, radioactivity, fuel fragmentation, cracking, oxidation, cladding strength and ductility, creep, and embrittlement	evaluate ATF fuel and cladding characteristics, such as form, size distribution, radioactivity, fuel fragmentation, cracking, oxidation, cladding strength and ductility, creep, and embrittlement	of water with uranium nitride fuel is more aggressive than uranium oxide fuels (Honnold et al, 2021); uranium silicide fuels: No disposal challenges or gaps have been identified

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10 CFR Part 71, *Packaging and Transportation of Radioactive Material*

10 CFR Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*

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40 CFR 261, *Identification and Listing of Hazardous Waste*

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APPENDIX A SPENT FUEL DATA

A.1 TRISO Fuel

The data in the below table was taken from the Nuclear Fuel Cycle Options Catalog (INL, 2022). The reference design is the General Atomic Modular High Temperature Gas Reactor. The Thermal efficiency is 50%. The fuel is 15.5% enriched Uranium oxycarbide fuel in the form of dense microspheres with a tristructural-isotropic coating. The tristructural-isotropic particles are bonded together into graphite compacts. The fuel compacts are then loaded into prismatic fuel blocks. The compacts are irradiated up to an average burnup of about 120 GWd/t. Discharged Fuel is stored and then sent to a disposal site. Pebble bed reactor technology was not considered here. Fission products with half-lives less than 1 year were removed from this list.

Table A-1 Isotopic composition of charged and discharged TRISO fuel for prismatic block type.

Isotope	Half Life (y)	ST1-charged (g)	ST1-discharged (g)
H3	12.32	0.00E+00	2.40E-08
SE79	2.95E+05	0.00E+00	1.68E+01
KR85	10.752	0.00E+00	7.08E+01
RB87	4.81E+10	0.00E+00	9.35E+02
SR90	28.9	0.00E+00	2.10E+03
ZR93	1.61E+06	0.00E+00	2.64E+03
NB94	2.03E+04	0.00E+00	2.25E-03
ZR96	2.35E+19	0.00E+00	2.97E+03
TC99	2.11E+05	0.00E+00	2.82E+03
MO100	7.30E+18	0.00E+00	3.38E+03
RU106	1.02	0.00E+00	3.55E+02
PD107	6.50E+06	0.00E+00	6.54E+02
CD108	1.90E+18	0.00E+00	1.25E-04
CD113	8.00E+15	0.00E+00	3.05E-01
CD114	2.10E+18	0.00E+00	4.04E+01
IN115	4.41E+14	0.00E+00	5.07E+00
CD116	3.30E+19	0.00E+00	1.49E+01
TE123	9.20E+16	0.00E+00	9.79E-03
SN124	1.20E+21	0.00E+00	2.70E+01
SB125	2.75856	0.00E+00	2.16E+01
SN126	2.30E+05	0.00E+00	6.27E+01
TE128	2.41E+24	0.00E+00	3.05E+02
I129	1.57E+07	0.00E+00	5.06E+02
TE130	3.00E+24	0.00E+00	1.33E+03
BA132	3.10E+21	0.00E+00	6.64E-05
BA133	10.5551	0.00E+00	9.35E-05
CS134	2.0652	0.00E+00	4.21E+02

Isotope	Half Life (y)	ST1-charged (g)	ST1-discharged (g)
XE134	5.80E+22	0.00E+00	5.52E+03
CS135	2.30E+06	0.00E+00	2.46E+03
XE136	2.40E+21	0.00E+00	6.95E+03
CS137	30.08	0.00E+00	4.41E+03
LA138	1.02E+11	0.00E+00	2.79E-02
CE142	5.00E+16	0.00E+00	4.16E+03
ND144	2.29E+15	0.00E+00	3.96E+03
PM147	2.6234	0.00E+00	5.39E+02
SM147	1.06E+11	0.00E+00	3.91E+02
SM148	7.00E+15	0.00E+00	5.27E+02
ND150	7.90E+18	0.00E+00	6.00E+02
EU151	1.70E+18	0.00E+00	8.74E-02
SM151	90	0.00E+00	5.78E+01
EU152	13.528	0.00E+00	1.86E-01
GD152	1.08E+14	0.00E+00	5.82E-03
EU154	8.601	0.00E+00	7.30E+01
EU155	4.753	0.00E+00	1.87E+01
GD160	3.10E+19	0.00E+00	3.19E+00
HO166M	1.20E+03	0.00E+00	3.12E-06
RA226	1600	0.00E+00	5.51E-07
AC227	21.772	0.00E+00	1.37E-07
TH228	1.9116	0.00E+00	3.56E-05
TH229	7932	0.00E+00	7.01E-06
TH230	7.54E+04	0.00E+00	2.37E-02
PA231	3.28E+04	0.00E+00	5.03E-03
TH232	1.40E+10	0.00E+00	1.61E-03
U232	68.9	0.00E+00	4.32E-03
U233	1.59E+05	0.00E+00	4.91E-03
U234	2.46E+05	3.10E+03	1.87E+03
NP235	1.0934	0.00E+00	2.78E-09
U235	7.04E+08	1.55E+05	4.86E+04
NP236	1.53E+05	0.00E+00	1.20E-03
PU236	2.858	0.00E+00	7.88E-10
U236	2.34E+07	3.24E+02	1.96E+04
NP237	2.14E+06	0.00E+00	1.77E+03
PU238	87.7	0.00E+00	7.66E+02
U238	4.47E+09	8.42E+05	7.82E+05
PU239	24110	0.00E+00	1.00E+04

Isotope	Half Life (y)	ST1-charged (g)	ST1-discharged (g)
PU240	6561	0.00E+00	4.57E+03
AM241	432.6	0.00E+00	1.91E+02
PU241	14.325	0.00E+00	4.58E+03
AM242M	141	0.00E+00	6.79E+00
PU242	3.75E+05	0.00E+00	2.22E+03
AM243	7370	0.00E+00	4.38E+02
CM243	29.1	0.00E+00	1.66E+00
CM244	18.1	0.00E+00	1.31E+02
PU244	8.00E+07	0.00E+00	8.18E-02
CM245	8423	0.00E+00	8.73E+00

A.2 High-Temperature Gas Reactors (HTGRs)

HTGR Radioactive Waste Streams

- Spent Nuclear Fuel
- Graphite/Carbon Waste
- Other radioactive Waste

HTGR Fuel

- Prismatic-block type: TRISO embedded in fuel compacts
- Pebble Bed: TRISO dispersed in a graphite pebble, basic unit for reactor core
- For some HTGR fuel designs, graphite up to 95% of fuel elements
- In image below, green blocks are treatment options, red block are waste forms with the likely waste classification in parentheses.

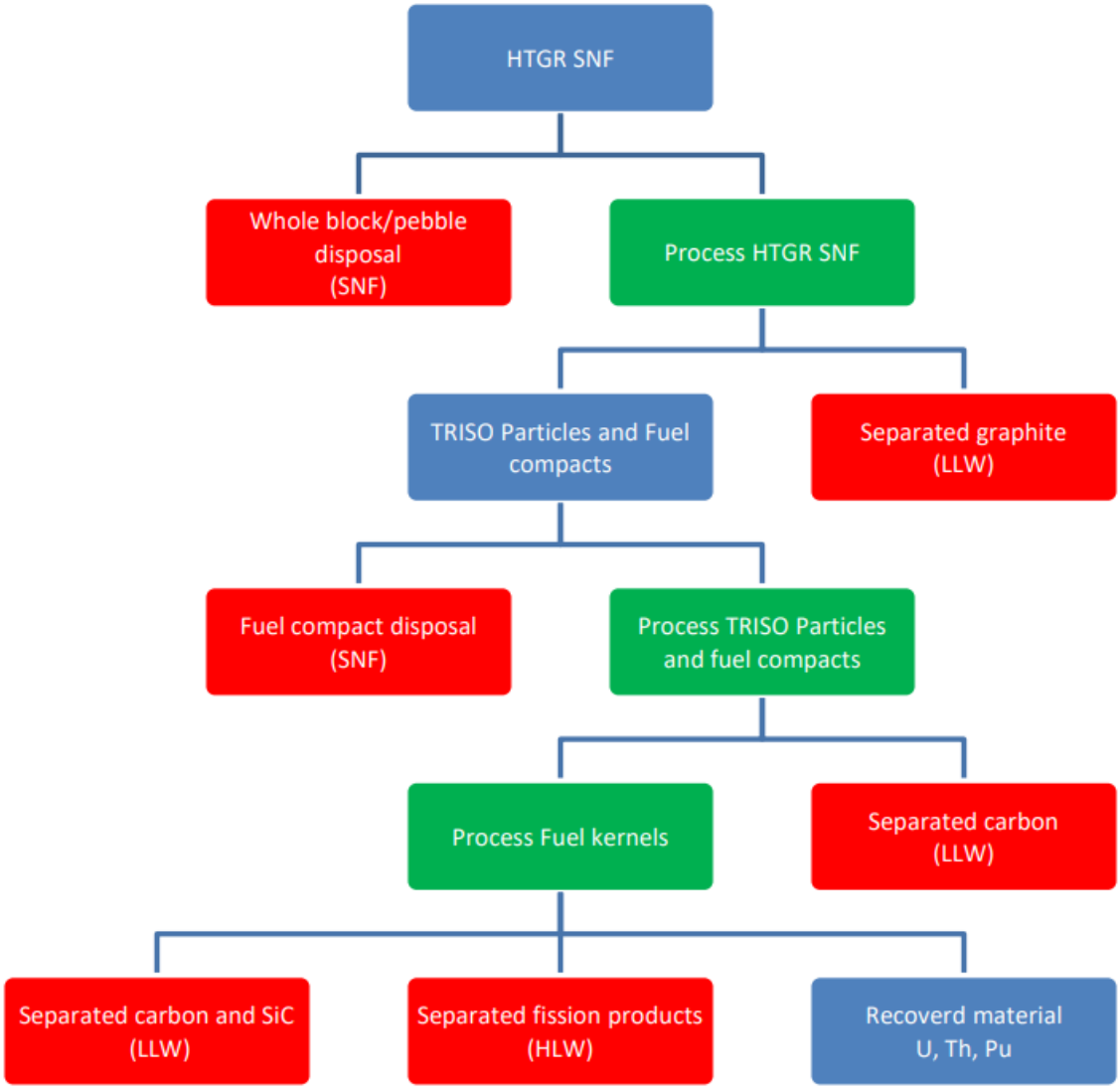

Figure A-1 HTGR SNF disposal pathways (Kitcher, 2020b).

Table A- 1 Experimental HTGRs.

	Peach bottom	**Dragon**	**AVR**	**HTTR**	**HTR-10**
Country	USA	UK	Germany	Japan	China
Operational Status	1967-1974, safe encl.	1968-1975 safe encl.	1967-1988 defueled	1998-xx in operation	2000-xx In operation
Electric Power MW(e)	40	--	15	--	--
Thermal Power MW(th)	115	20	46	30	10
Fuel Element type	pin	pin	spherical	Pin-in-block	Spherical
Power density (MWth/m3)	8.3	14	2.6	2.5	2
Coolant inlet/outlet temp	377/750	350/750	270/950	385/850 and 950	250/350/700/900
Mean He Pressure	2.5	2	1	4	3
Enrichment	HEU	HEU/LEU	HEU/LEU	LEU	LEU
Fuel	Carbide	Oxide	Carbide/Oxide	Oxide	Oxide
Coating	BISO	TRISO	BISO/TRISO	TRISO	TRISO
Pressure vessel	steel	steel	steel	steel	Steel

Table A- 2 Prototype HTGRs

	Fort St. Vrain	**THTR**
Country	USA	Germany
Operational Status	1976-1989 decommissioned	1986-1989 Safe encl.
Electric Power MW(e)	330	300
Thermal Power MW(th)	842	750
Fuel Element type	Prismatic	Spherical
Power density (MWth/m3)	6.3	6
Coolant inlet/outlet temp	405/784	270/750
Mean He Pressure	4.5	3.9
Enrichment	HEU	HEU
Fuel	Carbide	Oxide
Coating	TRISO	BISO
Pressure vessel	PCRV	PCRV

Table A-3 German commercial HTGR projects.

	PNP	HHT	HTR-500	HTR-Moduel	HTR-100
Electric Power MW(e)	--	500	500	80	100
Thermal Power MW(th)	500	1240	1250	200	258
Fuel Element type	spherical	Block/spherical	spherical	spherical	Spherical
Power density (MWth/m3)	4	5.5	7	3	3
Coolant inlet/outlet temp	300/950	440/850	280/700	250/750	250/740
Mean He Pressure	3.9	5	4.7	5	7
Enrichment	LEU	LEU	LEU	LEU	LEU
Fuel	Oxide	Oxide	Oxide	Oxide	Oxide
Coating	TRISO	TRISO	TRISO	TRISO	TRISO
Pressure vessel	PCRV	PCRV	PCRV	Steel	Steel

Table A- 4 Commercial HTGR projects in other countries.

	MHTGR	VGR-50	VGM-400	PBMR	GT/MHR	HTR-PM	HTR/VHTR (Antares)	NGNP (VHTR)	NGNP (VHTR)
Country	USA	Russia	Russia	SA	USA/Russia	China	France	USA	USA
Electric Power MW(e)	140	50	300	165	285	200	--	--	200
Thermal Power MW(th)	350	136	1060	400	600	2x250	600	600	500
Fuel Element type	prismatic	spherical	spherical	spherical	prismatic	spherical	prismatic	prsimatic	spherical
Power density (MWth/m3)	6	?	?	4.8	6.5	3.215	?	undecided	6.0
Coolant inlet/ outlet temp	319/685	296/810	350/950	500/900	510/850	250/700	400/1000	-/850-950	350/850-950
Mean He Pressure	9	4	5	9	7	7	5	undecided	9
Enrichment	LEU	HEU	LEU	LEU	U/Pu	LEU	LEU	LEU	LEU
Fuel	UCO	Oxide	Oxide	Oxide	Oxide	Oxide	UCO or UO2	UCO	UO2

Coating	TRISO	TRISO	TRISO	TRISO	TRISO	TRISO	TRISO	TRISO	TRISO
Pressure vessel	Steel	Steel	PCRV	Steel	Steel	Steel	Steel	Steel	Steel

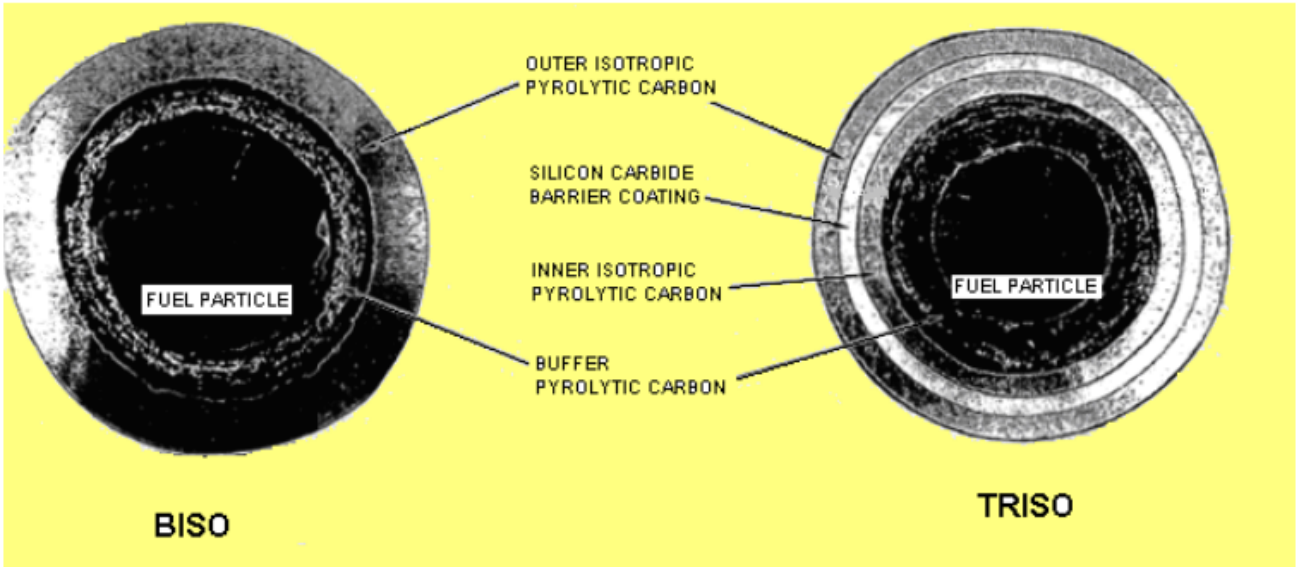

Figure A- 2 BISO versus TRISO image.

Most of the data in this section comes from: High temperature gas cooled reactors fuels and materials. IAEA, Vienna. March 2010. IAEA-TECDOC-1645-CD. ISBN 978-92-0-153110-2.

It's important to note that most of this information is from the past. Going forward, reactors are likely to use HALEU fuel rather than LEU or HEU.

A.3 Lead-cooled Fast Reactor Information

Table A- 5 Russian BREST reactor: Russia

Electric Capacity	300 MW(e)
Thermal Capacity	700 MW(th)
Thermal efficiency (%)	42
Core inlet/outlet Temps	420 C/540 C
Core diameter	2.6m
Core height	1.1m
Core Fuel	PuN-UN
Core configuration	Open Hexagonal
Max cladding temp	650 C
Core breeding ratio	~1
Enrichment	13.5%

Cladding	Chromium ferritic martensitic steel

Internally generated major actinides will be recycled at equilibrium content.

Table A- 6 European Lead Fast Reactor: Project funded by Sixth Framework Programme of EURATOM and coordinated by Ansaldo Nucleare.

Electric Capacity	600 MW(e)
Thermal Capacity	1500 MW(th)
Thermal efficiency (%)	42
Coolant temp	400 C/480 C
Core Diameter	4.5m
Core height	1.0m
Core fuel	pure MOX w/ depleted U and reactor grade Pu, or MOX fuel doped with equilibrium content of MAs
Core configuration	Wrapped Hexagonal
Max cladding temp	550 C
Core breeding ratio	~1

Recycling has been talked about, but not has not be detailed.

Table A- 7 Small Modular Natural Circulation Lead-Cooled Fast Reactor (SNCLFR-100): China (Guo *et al.*, 2021).

Electric Capacity	40 MW(e)
Thermal Capacity	100 MW (th)
Thermal efficiency (%)	
Coolant temp	400 C /480 C
Core Diameter	
Core height	3.4m (1m active core height)
Core fuel	MOX; Inner: PuO2 (16%) + UO2(84%) Middle: PuO2 (19%) + UO2 (81%) Outer: PuO2 (24%) + UO2(76%)
Core configuration	Closed octahedron array
Fuel pellet Diameter	9.8 mm
Cladding outer diameter	12.2 mm
Max cladding temp	

Information came from: Safety Analysis of Small Modular Natural Circulation Lead-Cooled Fast Reactor SNCLFR-100 Under Unprotected Transient by Chao Guo, Pengcheng Zhao, Jain Deng, and Hongxing Yu. June 8, 2021. Front. Energy Res., https://doi.org/10.3389/fenrg.2021.678939