GAP FINDINGS IN REGULATORY GUIDANCE ASSOCIATED WITH THE TRANSPORTATION AND DRY CASK STORAGE OF ADVANCED REACTOR FUEL TYPES

Prepared for

U.S. Nuclear Regulatory Commission
Contract No. 31310018D0001

Task Order No. 31310018F0113

Prepared by

George Adams
Yi-Ming Pan

Center for Nuclear Waste Regulatory Analyses
San Antonio, Texas

December 2021

ABSTRACT

This report documents potential gaps in regulatory guidance, along with information needed to address them, related to the transportation and dry cask storage of fresh and spent advanced reactor fuel (ARF). The focus of this report is on technical subjects covered in NRC guidance, and whether the guidance remains valid or if it should be modified to address specific technical issues involving transportation and dry cask storage of ARF. ARF technologies considered in this report include metal fuel for fast reactors and tristructural isotropic (TRISO) fuel for high-temperature gas-cooled reactors (HTGRs) and fluoride salt-cooled high-temperature reactors (FHRs). Information needs identified in this report generally refer to information to be developed and supplied by applicants, while gaps in regulatory guidance should be addressed by the NRC staff. Potential guidance gaps were found in the following areas of NUREG-2215 (NRC, 2020a) and NUREG-2216 (NRC, 2020b): (i) corrosion of non-fuel hardware, (ii) mechanical properties of cladding or coatings, and (iii) criticality safety. These gaps occur because current review guidance is associated with LWR UO_2 fuel enriched up to 5.0 weight percent ^{235}U; whereas many advanced reactor designs will use high-assay low-enriched uranium (HALEU), which has initial enrichment above 5 percent but below 20 percent and can achieve higher burnups. In addition, ARF does not have the same physical characteristics as traditional LWR fuel. For example, TRISO fuel can be in the form of spherical graphite matrix coated fuel particles where structural support and fission product retention is provided by the particle's SiC coating layer. In contrast, LWR UO_2 fuel pellets are contained within metal cladding for structural support and fission product retention. Public information is available for the transportation and storage of ARF; but additional studies are needed. For the corrosion of non-fuel hardware, information is needed on the types of non-fuel components and their material properties. For the mechanical properties of cladding or coatings, mechanical property models and mechanical property databases are needed. For criticality safety, evaluation codes need to be validated for ARF.

It is highlighted that this work is an assessment based on current publicly available information on the subject ARF technologies. The scope of the literature review was not exhaustive, particularly with respect to international sources. Additional information needs on technical issues affecting safety may be identified for specific technologies as more information is generated or becomes available. Additionally, this report did not extensively consider long-term aging effects associated with storage. Due to the wide variability in potential ARF technologies and fuel fabrication techniques, design-specific information provided early in any engagement process with the NRC would facilitate prompt identification of additional information needs specific to the candidate technology.

CONTENTS

ABBREVIATIONS/ACRONYMS

10 CFR	*Title 10 of the Code of Federal Regulations*
ARF	advanced reactor fuel
CNWRA®	Center for Nuclear Waste Regulatory Analyses
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor-II
EPRI	Electric Power Research Institute
FHR	fluoride salt-cooled high-temperature reactor
FRWG	Fast Reactor Working Group
HALEU	high-essay low-enriched uranium
HTGR	high-temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
KP	Kairos Power
LEU	low-enriched uranium
LWR	light water reactor
NEA	nuclear energy agency
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
SFR	sodium-cooled fast reactor
SNF	spent nuclear fuel
SRP	standard review plan
SiC	silicon carbide
TRISO	tristructural isotropic
UCO	Uranium Oxycarbide

ACKNOWLEDGMENTS

This report was prepared to document work performed by the Center for Nuclear Waste Regulatory Analyses (CNWRA®) for the U.S. Nuclear Regulatory Commission (NRC) under Contract No. 31310018D0001. The activities reported here were performed on behalf of the NRC Office of Nuclear Material Safety and Safeguards, Division of Fuel Management. The report is an independent product of CNWRA and does not necessarily reflect the views or regulatory positions of the NRC.

The authors thank Osvaldo Pensado for his technical review and David Pickett for his programmatic review. The authors also thank Arturo Ramos for providing formatting and word processing support in preparation of this document.

QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

DATA: There are no original CNWRA-generated data in this report. Sources of other data should be consulted for determining the level of quality of those data.

ANALYSES AND CODES: No codes were used in the analyses contained in this report.

1 INTRODUCTION

1.1 Background

To help prepare for regulatory interactions and potential license applications for managing fresh and spent fuels for non-light water reactor (LWR) technologies, the U.S. Nuclear Regulatory Commission (NRC) staff seeks to identify potential regulatory guidance gaps in NUREG-2215 (NRC, 2020a) and NUREG-2216 (NRC, 2020b). In addition to identifying potential gaps, this report identifies information and technical analyses required to address these gaps. Potential regulatory guidance gaps are associated with the transportation of fresh advanced reactor fuel (ARF) and the transportation and dry cask storage of spent ARF. Potential ARF types that are addressed in this report include metal fuel for sodium-cooled fast reactors (SFRs) and tristructural isotropic (TRISO) fuel for high-temperature gas-cooled reactors (HTGRs) and fluoride salt-cooled high-temperature reactors (FHRs). Early identification of potential regulatory guidance gaps can facilitate the development of information and technical analyses needed to address these gaps.

1.2 Purpose and Scope

The Center for Nuclear Waste Regulatory Analyses (CNWRA®) has been tasked with reviewing publicly available literature and applicable NRC guidance to identify potential information needs and regulatory guidance gaps relevant to the transportation and dry cask storage of ARF. The ARF types considered for this report are metal fuel and TRISO fuel enriched to less than 20 weight percent ^{235}U[1]. Specifically, high-assay low-enriched uranium (HALEU) is considered for advanced reactor designs because smaller fuel assemblies are possible and higher burnups can be achieved. HALEU has a ^{235}U assay (or concentration) greater than 5 percent and less than 20 percent. The following NRC regulations and guidance were consulted:

- *Title 10 of the Code of Federal Regulations* (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;
- NUREG-2215, Standard Review Plan (SRP) for Spent Fuel Dry Storage Systems and Facilities (NRC, 2020a); and
- NUREG-2216, SRP for Transportation Packages for Spent Fuel and Radioactive Material (NRC, 2020b).

The scope of this report includes (i) identifying potential regulatory guidance gaps and (ii) determining the information and technical analyses needed to address these gaps. Appendix I documents the gap/needs analysis for transportation of fresh ARF, Appendix II documents the gap/needs analysis for storage of spent ARF, and Appendix III documents the gap/needs analysis for transportation of spent ARF. Section 2 of this report describes the potential regulatory guidance gaps identified from the information needs and gap analyses presented in detail in the appendices. Section 3 describes information that currently is available related to the identified gaps along with any additional information and technical analyses that

[1]Nuclear fuel enriched to less than 20 weight percent ^{235}U is low-enriched uranium (LEU) fuel (See 10 CFR 50.2).

may be needed. This report is not aimed at identifying deficiencies in the current regulations, but instead it discusses potential information needs and regulatory guidance gaps in the SRPs for light water reactor fuels (NRC, 2020a and 2020b).

2 RESULTS FROM GAP ANALYSIS

Appendices I, II, and III tabulate the results from an information needs and gap analysis for transportation and storage of advanced reactor fuel (ARF). The focus of this section is on potential regulatory guidance gaps found in NUREG-2215 (NRC, 2020a) and NUREG-2216 (NRC, 2020b). The gaps relate to (i) corrosion of non-fuel hardware, (ii) mechanical properties of cladding or coatings, and (iii) criticality safety. The gaps are discussed separately in this section.

2.1 Corrosion of Non-Fuel Hardware

Regulatory Guidance: Section 8.5.13.2 of NUREG-2215 (NRC, 2020a) and Section 7.4.10.2 of NUREG-2216 (NRC, 2020b)

Relevant Gap Tables: Appendix I, Tables I-6 and I-12; Appendix II, Tables II-5 and II-11; and Appendix III, Tables III-6 and III-12

The relevant guidance in SRPs addresses corrosive reactions between transportation package and storage container components and their contents. The guidance is specific to certain non-fuel hardware components[1] encased in zirconium alloy and stainless steel, where the transportation or storage canister is made of stainless steel with stainless steel or aluminum basket components. Regarding advanced reactors, non-fuel hardware components may be different from those used in light water reactors (LWRs), and these components may use materials other than the zirconium alloy and stainless steels previously analyzed for LWRs and referred to in current SRPs. Because of these differences in non-fuel hardware components and materials, information needs arise in the technical bases available to staff for making a regulatory conclusion, and related gaps in the SRP guidance can be identified and described. For example, high-temperature gas-cooled reactors (HTGRs) and fluoride salt-cooled high-temperature reactors (FHRs) use graphite components. Furthermore, NUREG-2215 (NRC, 2020a) and NUREG-2216 (NRC, 2020b) address wet loading and dry storage environments associated with LWRs, but the environment can be different for advanced reactors. For example, for FHRs, the moderator is a fluoride salt, which could be present in solid form on the internal components of a transportation package or storage container. Fluoride salt in solid form produces fluorine gas under radiolysis, which is a strong oxidizer. Thus, the internal components of a transportation package or storage container could be subjected to corrosion processes that result from the fluoride salt environment; the current SRPs are not sufficiently general to address degradation of materials by fluorine.

2.2 Mechanical Properties of Cladding or Coatings

Regulatory Guidance: Section 8.5.15.2 of NUREG-2215 (NRC, 2020a) and Sections 7.4.13 and 7.4.14.2 of NUREG-2216 (NRC, 2020b)

Relevant Gap Tables: Appendix I, Tables I-6 and I-12; Appendix II, Tables II-5 and II-11; and Appendix III, Tables III-6 and III-12

[1]Non-fuel hardware for LWRs includes neutron source assemblies, burnable poison rod assemblies, thimble plug devices, and other hardware associated with reactivity, thermal, and dose control.

The relevant guidance provides criteria for examining the mechanical properties of cladding materials. The guidance addresses aluminum, stainless steel, and zirconium alloy cladding materials credited with maintaining the structural integrity of LWR fuel during transportation, loading, and storage.

Advanced[2] stainless steel cladding materials (i.e., D9[3] and HT9[4]) were developed for sodium-cooled fast reactors (SFRs) to accommodate the higher temperatures and more severe radiation environment in these reactors. These advanced cladding materials and the metal fuel itself need to be evaluated for structural integrity during transportation, loading, and storage. Whereas LWR fuel relies solely on cladding for structural support, the metal fuel and cladding together may be credited for structural support of SFR fuel. Because the SRPs provide guidance specific to LWR cladding, there is a potential regulatory guidance gap regarding advanced cladding materials and other systems providing structural integrity.

Tristructural isotropic (TRISO) fuel particles used in HTGRs and FHRs do not have cladding. Rather, TRISO fuel particles rely on a silicon carbide (SiC) layer for structural support and fission product retention. The SiC-coated[5] TRISO fuel particles are embedded in a cylindrical or spherical graphite matrix. The potential regulatory guidance gap is related to the observation that unlike LWR fuel elements where the UO_2 fuel pellets are surrounded by metal cladding for structural support, the graphite matrix is not contained within another support structure, and a ceramic coating layer–not cladding–is credited with structural support and fission product retention. Therefore, the SiC fuel coating layer and graphite matrix mechanical properties need to be evaluated for structural integrity during transportation, loading, and storage, which are not covered by existing guidance in the SRPs. Furthermore, NUREG-2216 (NRC, 2020b) provides guidance to ensure that the fuel is maintained in an analyzed configuration during transportation. Therefore, the graphite matrix and supporting structures within the transportation canister need to be evaluated for structural integrity if credited for configuration control during transportation; however, this aspect is not addressed in the existing guidance.

2.3 Criticality Safety

Regulatory Guidance: Sections 7.5.5.1, 7.5.5.3, and 7.5.5.4 of NUREG-2215 (NRC, 2020a) and Sections 6.4.7.1, 6.4.7.3, and 6.4.7.4 of NUREG-2216 (NRC, 2020b)

Relevant Gap Tables: Appendix II, Tables II-4 and II-10 and Appendix III, Tables III-5 and III-11

The relevant guidance addresses burnup credit criticality evaluation of LWR UO_2 fuel irradiated to an assembly-average burnup of 60 GWd/MTU. The guidance is applicable to low enriched uranium (LEU) fuel with up to 5 percent ^{235}U. A potential regulatory guidance gap exists because many advanced reactors will use high-assay low-enriched uranium (HALEU) fuel with a higher initial enrichment from 5 percent to 20 percent. This higher initial enrichment is not addressed in current SRPs. In addition, HALEU could enable higher burnups than 60 GWd/MTU to be achieved, potentially up to 210 GWd/MTU (NEA, 2014). Current analysis methods consider actinide and fission product compositions in irradiated UO_2 fuel, but the compositions of ARFs are different. For example, TRISO fuel may be uranium oxycarbide

[2]The term “advanced” is referring to cladding materials such as D9 and HT9 and new materials that are designed to accommodate the higher temperatures and more severe radiation environments of SFRs.
[3]D9 is a titanium modified variant of Type 316 stainless steel (Makenas, 1986).
[4]HT9 is a 12 percent chromium, 1 percent molybdenum ferritic/martensitic stainless steel (Brown et al., 2012).
[5]In addition to the SiC coating layer there are other pyrocarbon coating layers surrounding the fuel kernel in TRISO fuel.

(UCO)[6], and metal fuel may be uranium alloyed with zirconium. Therefore, the irradiated compositions also will be different than irradiated UO_2. Additionally, the guidance requires validated codes for burnup credit criticality evaluations (i.e., codes for isotopic depletion and k_{eff} calculations). Currently available codes and analysis methods would not be generally valid at higher initial enrichments, higher burnups, and different initial fuel compositions.

[6]Uranium oxycarbide (UCO) is a short-hand notation for a mixture of uranium dioxide (UO_2) and uranium carbide.

3 INFORMATION OR TECHNICAL ANALYSES NEEDED TO ADDRESS POTENTIAL GAPS IN REGULATORY GUIDANCE

The potential regulatory guidance gaps identified in Section 2 are discussed in this section along with technical analyses and information needed to address the gaps. Gaps related to the corrosion of non-fuel hardware and mechanical properties are specific to the advanced reactor fuel (ARF) type [i.e., metal or tristructural isotropic (TRISO)]. Therefore, these gaps are discussed separately for each type of fuel. However, the gap related to criticality safety is not specific to the ARF type, so that discussion applies to both metal and TRISO fuel.

Publicly available information was searched in an effort to find information or analyses relevant to each of the regulatory guidance gaps. Although the information is extensive, information could not be found to address all of the gaps. Therefore, information that is currently available is discussed for each gap along with any additional information and technical analyses that may be required. The discussion recognizes that advanced reactor designs continue to evolve, designers and fabricators continue to develop additional information and provide further analyses, and the scope of the literature review was not exhaustive (particularly with respect to international sources). Therefore, the term "may" generally is used in identifying information and analyses associated with the current assessment of gaps.

3.1 Corrosion of Non-fuel Hardware

3.1.1 Metal Fuel

Metal fuel elements used in sodium-cooled fast reactors (SFRs) are designed with a metallic fuel slug sodium bonded to stainless steel cladding. The Aurora advanced reactor metal fuel is uranium alloyed with 10 percent zirconium (U-10Zr) (Oklo, 2020). The addition of zirconium increases the fuel melting point and minimizes fuel/cladding chemical interaction (Carmack et al., 2009; Sofu, 2019). The fuel element has a gas plenum to collect fission product gases. During irradiation, fission gases are produced forming voids in the fuel and causing the fuel to swell. Eventually these voids interconnect, releasing fission gases to the plenum. SFRs use low-pressure liquid sodium as the reactor coolant. The oxygen-free and water-free sodium environment prevents corrosion while the fuel is in the reactor, but sodium reacts chemically with air and water.

Non-fuel hardware included in transportation and storage canisters can include components associated with reactivity, dose, and thermal control. Corrosive conditions can degrade the canister's structural integrity, so the current review guidance addresses corrosive reactions between the canister and non-fuel hardware components. Oklo (2020) describes non-fuel hardware that includes absorber rods and shutdown rods made of Type 316L stainless steel filled with boron carbide (B_4C). This type of control element is consistent with the regulatory guidance[1]. However, other materials also may be used for reactivity control. For example, Guo et., al (2020) and Guo (2018) describe the advantages of europium oxide (Eu_2O_3) and gadolinium oxide (Gd_2O_3) control rods in comparison to B_4C rods. Guo et., al (2020) also describe the use of burnable poisons consisting of neptunium, depleted B_4C, and zirconium hydride ($ZrH_{1.62}$). Non-fuel hardware components are expected to continue to evolve as advanced reactor designs are developed. This new hardware may differ from the zircaloy or

[1]Section 8.5.13.2 of NUREG-2215 (NRC, 2020a) and Section 7.4.10.2 of NUREG-2216 (NRC, 2020b) describe stainless steel clad control elements containing boron carbide.

stainless steel clad components described in the review guidance. Therefore, potential regulatory guidance gaps exist in Section 8.5.13.2 of NUREG-2215 (NRC, 2020a) and Section 7.4.10.2 of NUREG-2216 (NRC, 2020b), given their focus on specific materials such as boron carbide clad in stainless steel or zircaloy. Because of the uncertainty in future designs, development of technology-neutral guidance is desirable. Information needs and analyses to address these gaps include the following.

- The material composition and design of non-fuel hardware components.
- Analyses for chemical, galvanic, or other reactions, including potential implications of damaged non-fuel hardware components.

3.1.2 TRISO Fuel

TRISO fuel elements are in the form of coated particles embedded in a cylindrical graphite matrix for prismatic cores or a spherical matrix for pebble bed cores. These two fuel element forms may be used in either high-temperature gas-cooled reactor (HTGR) designs or fluoride salt-cooled high-temperature reactor (FHR) designs. Whereas the HTGR uses helium as a coolant, the FHR uses a liquid fluoride salt coolant. The FHR design combines the graphite matrix coated particle fuel and graphite moderator from the HTGR with a liquid fluoride salt coolant (Forsberg and Peterson, 2015). This coolant can provide some design improvements because liquids are better coolants than gases. Also, fluoride salts are compatible with graphite-based fuels.

In the FHR design, solid salt coolant may remain on the SNF graphite matrix and non-fuel hardware after loading in a transportation or storage container. The solid salt can undergo radiolysis and generate fluorine gas, which decomposes to hydrofluoric acid on contact with moisture. Fluorine is the most powerful oxidizer known (Perdue University, 2021), so corrosion of non-fuel hardware inside a storage or transportation canister is possible if solid fluoride salt is present.

The current regulatory guidance addresses corrosion of light water reactor (LWR) non-fuel metallic hardware and considers the wet loading/dry storage environment for LWR SNF. In contrast, HTGRs and FHRs use graphite components. Also, the storage environment can be different because materials removed from FHRs may have solid residual salt on their surfaces. As described for the FHR demonstration reactor design, some non-fuel hardware that may be used in a transportation or storage canister are molybdenum-hafnium-carbide alloy control rods and graphite burnable absorber rods having small weight fractions (<2 percent) of natural B_4C (Qualls et al., 2016). Gougar (2014) describes metal-clad control rods made of Incoloy-800H (nickel-iron-chromium alloy) tubes enclosing B_4C annular graphite compacts as well as absorbers made of pyrolytic carbon-coated spheres of B_4C granules embedded in a graphite matrix. These non-fuel hardware components differ in design from the zircaloy or stainless steel clad components described in the current review guidance. Because non-fuel hardware components will continue to evolve as advanced reactor designs are developed, a technology-neutral approach to guidance is desirable. Therefore, potential regulatory guidance gaps exist in Section 8.5.13.2 of NUREG-2215 (NRC, 2020a) and Section 7.4.10.2 of NUREG-2216 (NRC, 2020b) given their focus on zircaloy or stainless steel clad control elements. Information needs and analyses to address these gaps include the following.

- The material composition and design of non-fuel hardware components.
- Analyses for chemical, galvanic, or other reactions, including potential implications of damaged non-fuel hardware components.
- The potential for solid fluoride salt being present, including the potential for generating highly corrosive fluorine from radiolysis.

3.2 Mechanical Properties of Cladding or Coatings

3.2.1 Metal Fuel

Section 2 highlights potential regulatory guidance gaps when evaluating the mechanical properties of cladding materials used in metal fuels. For LWRs, fuel cladding can provide structural support to the fuel element and serve as a barrier to fission product release during transportation, loading, and storage. It is typically credited with maintaining uncanned[2] spent fuel in its analyzed configuration. While metal fuel elements used in SFRs also rely on the cladding as a barrier to fission product release and for structural support, the metal fuel elements might also rely on the metal fuel itself for structural support.

Current review methods provide guidance for examining the mechanical properties of zirconium and aluminum alloy cladding. Although the current guidance provides some reference to stainless steels used as cladding in LWRs, it does not reference more advanced[3] stainless steel alloys used in advanced reactor metal fuel designs such as the Oklo reactor. [See for example the license application of an advanced micro-reactor submitted to NRC (Oklo, 2020).]

Raj et al. (2009) describe developments in cladding materials for SFRs in India, stating

> *Materials in sodium-cooled fast reactors need to be capable of operating at higher temperatures in a more severe radiation environment as compared to materials in thermal nuclear reactors. The presence of sodium presents additional challenges to maintain and monitor low levels of oxygen and nitrogen dissolved in the liquid sodium. Thus the challenges for core components in sodium-cooled fast reactors revolve around radiation resistance, high-temperature mechanical properties and chemical compatibility with the fuel as well as the liquid sodium coolant.*

Raj et al. (2009) state that austenitic stainless steels are favored for fuel pin cladding and other core components because of their strength up to 923 K [1,202 °F]. The current limitation on fuel burn-up is void swelling of core structural materials. Compared to austenitic stainless steels, ferritic steels (e.g., HT9) have greater void swelling resistance, but they have poorer tensile and creep strengths at temperatures above approximately 873 K [1,112 °F].

Stainless steel cladding materials for metal fuel include Type 304L, Type 316, D9 (15%Cr-15%Ni-0.2%Ti) (a titanium modified variant of Type 316 stainless steel), and HT9

[2] Uncanned spent fuel is undamaged or intact spent fuel that is not placed inside a separate fuel can within the dry storage system.

[3] Advanced stainless steel cladding materials such as D9 and HT9 were developed for their greater void swelling resistance in the higher irradiation environment of an SFR.

(a high strength ferritic-martensitic stainless steel alloy) [Fast Reactor Working Group (FRWG), 2018]. Type 316 has been used instead of Type 304L because it has better swelling resistance and experiences reduced fuel/cladding interdiffusion. Interdiffusion reduces the load-bearing thickness of the cladding wall (Crawford et al., 2007). D9 is a further improvement; exhibiting greater swelling resistance compared to cold worked Type 316 stainless steel (Makenas, 1986). In addition, Hackett and Povirk (2012) found HT9 has excellent swelling resistance to doses above 200 displacements per atom. They also describe a substantial irradiation effects database on mechanical properties (i.e., Maloy et al., 2011 and Maloy et al., 2006). Carmack et al. (2009) state that future cladding designs may use a ferritic-martensitic stainless steel where fine oxide powder is dispersed to improve high-temperature strength and stress rupture properties. Garner (1993) and IAEA (2012b) provide additional data on metal fuel cladding materials.

Stainless steel cladding performance in storage environments may be challenged by sensitization, intergranular attack, stress corrosion cracking, thermal aging, and radiation embrittlement (Alexander and Nanstand, 1995; Chandra et al., 2012; Guenther et al., 1996). Sensitization involves the formation of chromium carbides at grain boundaries, which depletes chromium in areas adjacent to the boundaries. These depleted zones can experience intergranular attack and intergranular stress corrosion cracking. Sodium-bonded spent metal fuel may experience degradation during storage, particularly oxidation, hydriding, fragmentation, and restructuring-swelling (Guenther et al., 1996).

Section 8.5.15.2 of NUREG-2215 (NRC, 2020a) and Sections 7.4.13 and 7.4.14.2 of NUREG-2216 (NRC, 2020b) provide guidance for examining the mechanical properties of cladding materials to ensure structural integrity of the fuel element based on LWR technology. Whereas LWR fuel relies solely on cladding for structural support, the metal fuel and cladding together may be credited for structural support of SFR fuel elements. The difference in how structural support is accomplished in metal fuel creates a potential regulatory guidance gap. In order to address this gap, the following Information needs and analyses are identified.

- The cladding and metal fuel itself may be credited with structural support of the metal fuel element, with sufficient justification. Consequently, testing and modeling can demonstrate the fuel element remains intact from challenges during transportation, loading, and storage.

- The NRC has determined that mechanical property models such as PNNL-17700 (Geelhood et al., 2008) are acceptable. However, similar models may not exist for newer cladding materials used in metal fuel designs. Therefore, additional mechanical property models may need to be developed or existing models may need to be updated.

- There may be limited data on new cladding materials and some properties may be dependent on manufacturing processes, so manufacturer test data are needed to evaluate the mechanical properties of new cladding materials.

- Codes and standards are approved sources of cladding material data; however, codes and standards may not be available for new cladding materials. Therefore, codes and standards may need to be developed or updated for any new cladding materials.

- Metal fuel elements may experience significant swelling during operations. In addition, metal fuel elements may experience degradation during storage. Therefore, structural analyses may be needed to ensure the spent metal fuel element has adequate structural integrity to withstand the additional stresses experienced during transportation, loading, and storage.

- Limited data may be available on the mechanical properties of cladding materials for metal fuel and the metal fuel itself, especially when these materials are used in a high temperature, high burnup, and high irradiation environment. Also, the mechanical properties can change over time due to age-related degradation phenomena. Therefore, a materials properties database may be needed to account for the effects of temperature, irradiation, burnup, and aging on metal fuel and its cladding.

3.2.2 TRISO Fuel

Section 2 of this report highlights potential regulatory guidance gaps when evaluating the mechanical properties of TRISO fuel. Current review methods provide guidance for examining the mechanical properties of cladding for UO_2 fuel used in LWRs. The cladding provides structural support for the fuel and containment of fission products. However, the TRISO fuel particle does not rely on cladding for structural support; rather, it relies on a SiC coating that surrounds it. In addition, the TRISO particles are embedded in a graphite matrix. The graphite matrix does not have an external support structure such as metal cladding, and graphite is susceptible to failure. For example, Marsden (2001) states that graphite component physical properties are significantly changed by irradiation, and stresses generated following irradiation can lead to component failure.

TRISO fuel is used primarily in HTGRs having graphite cores and cooled by pressurized helium with outlet temperatures between 700 and 950 °C [1,292 and 1,742 °F]. It is characterized by statistically low coating failure fractions and good fission product retention under extreme conditions such as temperatures of 1,600 °C [2,912 °F] for hundreds of hours (Demkowicz et al., 2018). The fuel particles are designed to withstand high internal gas pressure without releasing their fission products. The TRISO fuel particle consists of a 350 to 600 μm [13,780 μin to 23,622 μin] uranium-bearing kernel. The kernel consists of fissile or fertile material usually in the form of UO_2, PuO_2, or UCO with enrichments from 8 to 20 percent (IAEA, 2010). The kernel is surrounded by four layers. The first layer is a buffer layer: a porous pyrocarbon layer providing void space for fission gases. The second layer is an inner pyrocarbon layer that protects the kernel and helps retain some fission gases. The third layer is a SiC layer that serves as the primary pressure boundary (IAEA, 2010) providing containment for both gaseous and non-gaseous fission products and structural strength for the particle. It also is very resistant to chemical attack (Del Cul et al., 2002). The fourth and outer layer is a pyrocarbon layer that protects the SiC layer and provides an additional barrier to fission product release. The TRISO-coated particles have an overall diameter in the range of 500 to 1,000 μm [19,685 μin to 39,370 μin] (IAEA, 2010).

Marschman et al. (1993) describe a failure mechanism for the TRISO fuel. The fuel kernel used at Fort St. Vrain was carbide-based instead of oxide-based or oxycarbide-based. Lanthanide fission products were produced which reacted with the SiC layer causing failure of the fuel particle. Marschman et al. (1993) determined if at least 15 percent of the carbide is converted to oxide, then lanthanides are retained in the kernels preventing their reaction with the SiC layer. Therefore, newer TRISO fuel designs are oxide-based or oxycarbide-based to allow lanthanide fission products to be retained in the kernel.

TRISO-coated particles are made into fuel elements. There are currently two different designs for fuel elements, but the TRISO-coated particle is essentially the same for each. One design is a hexagonal block fuel element, and it is used in prismatic cores; the other fuel element design is spherical, and it is used in pebble-bed cores. For prismatic fuel block cores, the TRISO fuel particles are formed into compacts, which are loaded into predrilled holes in a graphite block to make the fuel element. Del Cul et al. (2002) describe the compacts for a modular helium-cooled reactor as right-circular cylinders 12.7 mm in diameter by 49.3 mm in length [0.5-in diameter by 1.94-in long]. The fuel elements are hexagonal prisms machined from graphite and the compacts are inserted one on top of the other into fuel holes machined in the hexagonal fuel elements (IAEA, 2010). The second type of HTGR is the pebble bed reactor. IAEA (2010) describes a modular pebble bed reactor in which the fuel elements (or pebbles) are 60 mm-diameter [2.37 in-diameter] spheres formed by embedding coated fuel particles in a graphite matrix. The pebble bed modular reactor (PBMR) contains about 452,000 pebbles (IAEA, 2012a). It allows on-power refueling where pebbles move gradually through the core from top to bottom. Once they exit the core, pebbles are assessed for burnup. If burnup is insufficient, they are returned to the top of the core. In addition to the PBMR, advanced reactor designs currently in various stages of development that utilize pebble style fuel include the X-energy (2021) Xe-100 reactor and Kairos Power (KP) (2021) KP-FHR.

TRISO fuel does not rely on cladding for structural support or containment of fission products. Instead, the SiC layer is the primary structural component of TRISO fuel. This layer also provides the radionuclide retention capabilities for TRISO-coated particles. The effectiveness of the SiC layer as a barrier to fission product release within the TRISO fuel is critical to overall fuel performance. However, the current regulatory guidance is designed for LWR fuels that rely on cladding for structural support and fission product retention. The TRISO fuel particles are contained within a graphite matrix and the graphite matrix is not contained within another support structure such as metal cladding. Therefore, potential regulatory guidance gaps exist in Section 8.5.15.2 of NUREG-2215 (NRC, 2020a) and Sections 7.4.13 and 7.4.14.2 of NUREG-2216 (NRC, 2020b) because the SiC fuel coating layer and graphite matrix mechanical properties need to be evaluated for structural integrity during transportation, loading, and storage. Also, NUREG-2216 (NRC, 2020b) provides guidance to ensure the fuel is maintained in an analyzed configuration during transportation. Therefore, the graphite matrix and supporting structures within the transportation canister need to be evaluated for structural integrity if credited for configuration control. Information needs and analyses to address these gaps are as follows.

- The SiC layer is the primary barrier to fission product release, and it provides structural support for the fuel particle. Fuel particles are embedded in a cylindrical or spherical graphite matrix. The TRISO fuel needs to be maintained in an analyzed configuration during transportation and storage. Testing and modeling may show that the TRISO fuel remains in an analyzed configuration and demonstrate the graphite matrix remains intact from challenges during transportation, loading, and storage.

- The NRC has determined mechanical property models such as PNNL-17700 (Geelhood et al., 2008) are acceptable for examining the properties of metal used as a cladding material. However, similar models may not exist for the coating layers used for structural support and fission product retention in TRISO fuel. Therefore, additional mechanical property models may need to be developed for TRISO fuel.

- There may be limited data on the coating layers for TRISO fuel particles and limited data on the fuel particles once they are embedded in a cylindrical or spherical graphite matrix. Also, some properties may be dependent on manufacturing processes, so manufacturer test data are needed to evaluate the mechanical properties of the TRISO fuel coating layers and graphite matrix.

- Degradation of TRISO fuel during transportation, loading, and storage has not been reported, and limited information is available for spent TRISO fuel stored at the Fort St. Vrain ISFSI (DOE, 2010, 1992; Marschman et al., 1993). Therefore, structural analyses may be needed to ensure the fuel particle, graphite matrix, and supporting structures in transportation and storage canisters have adequate structural integrity to withstand the additional stresses experienced during transportation, loading, and storage.

- Limited data may be available on the mechanical properties of the TRISO fuel coating layers. Also, the mechanical properties can change over time due to age-related degradation phenomena. Therefore, a materials properties database may be needed to account for the effects of temperature, irradiation, burnup, and aging on TRISO coatings.

3.3 Criticality Safety

HALEU has a ^{235}U assay (or concentration) above 5 percent but below 20 percent. Whereas existing LWRs operate on low enriched uranium (LEU) with up to 5 percent ^{235}U, many advanced reactor designs will use HALEU because the fuel assemblies and reactors can be smaller, the reactors do not need to be refueled as often, and higher burnups can be achieved (Centrus, 2021).

Although HALEU is expected to be fabricated into both metal and TRISO fuel for advanced reactors, it is not yet commercially produced (NEI, 2018a). Centrus and X-energy are developing a fuel fabrication facility for UCO TRISO fuel (Centrus, 2021). In addition, Centrus is working with TerraPower to produce HALEU and fabricate it into metal fuel assemblies for the Natrium demonstration reactor (World Nuclear News, 2020). In the meantime, current production is limited to small amounts of HALEU fabricated by blending down U.S. government stocks of high-enriched uranium (NEI, 2018b).

There are technical challenges related to demonstrating criticality safety for advanced reactor fuels using HALEU. For example, there have been very few criticality benchmark experiments for enrichments between 5 percent and 20 percent. Jarrell (2018) describes over 5,000 approved International Criticality Safety Benchmark Evaluation Project criticality benchmarks but only 376 were in the range between 5 and 25 percent. Furthermore, the applicability of benchmarks is not solely dependent on enrichment but also must take materials, configuration and design into account.

Applicants have several means to demonstrate criticality safety. First, additional criticality benchmark experiments could be conducted to expand the number of available experiments for validating HALEU systems. Second, sensitivity/uncertainty analysis techniques could be used to demonstrate sufficient existing experiments are applicable to HALEU systems. Third, applicants could include enough margin in the criticality analysis for HALEU systems to account for validation uncertainties due to insufficient criticality experiments.

Section 2 highlights potential regulatory guidance gaps associated with storing and transporting high burnup fuel with higher initial enrichments. Current review methods provide guidance associated with LWR UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated to an assembly-average burnup value not exceeding 60 GWd/MTU. However, TRISO fuel used in modern high-temperature gas-cooled reactor (HTGR) designs may experience a fuel burnup of 150–210 GWd/MTU (NEA, 2014) and EBR-II experiments have shown metal fuel burnup between 38 and 143 GWd/MTU [Fast Reactor Working Group (FRWG), 2018]. These enrichment and burnup parameters may require changes to the current regulatory guidance limits defined for transporting and storing LWR SNF. [See Section 7.5.5.1 of NUREG-2215 (NRC, 2020a) and Section 6.4.7.1 of NUREG-2216 (NRC, 2020b) for regulatory guidance limits.] In addition, the current regulatory guidance for burnup credit criticality evaluations ensures validated codes and methods for performing isotopic depletion and k_{eff} calculations. Current analysis methods consider actinide and fission product compositions in UO_2 fuel irradiated to a maximum 60 GWd/MTU burnup. The compositions of ARFs are different, and the irradiated compositions at higher burnups also will be different. Therefore, potential regulatory guidance gaps exist in Sections 7.5.5.1, 7.5.5.3, and 7.5.5.4 of NUREG-2215 (NRC, 2020a) and Sections 6.4.7.1, 6.4.7.3, and 6.4.7.4 of NUREG-2216 (NRC, 2020b). Information needs and analyses to address these gaps are as follows.

- Burnup credit analysis methods may need to be developed for HALEU ARF, if burnup credit is desired. Current burnup credit analysis methods consider actinide and fission product compositions in LWR UO_2 fuel, but the compositions of ARFs are different. For example, it may be metal fuel composed of uranium alloyed with zirconium or TRISO fuel composed of UCO. Therefore, current burnup credit analysis methods may not be applicable to HALEU fuel used in advanced reactors operated to higher burnups than LWRs.

- Isotopic depletion computer codes need to be validated to account for the irradiation environment, geometry, and configuration of HALEU ARF.

- Destructive radiochemical assay data needs to be compiled for HALEU fuel and applied to validate computer codes used to calculate isotopic depletion, for those applications desiring burnup credit.

- Codes need to be validated for criticality (k_{eff}) calculations.

- Additional criticality benchmark experiments may need to be conducted to support criticality code validation of HALEU ARF at higher burnups.

4 SUMMARY AND CONCLUSIONS

This report identifies potential regulatory guidance gaps associated with the review and certification of transportation packages and dry storage casks for fresh and spent advanced reactor fuel (ARF). ARF technologies considered in this report are metal fuel used in sodium-cooled fast reactors (SFRs) and tristructural isotropic (TRISO) fuel used in high-temperature gas-cooled reactors (HTGRs) and fluoride salt-cooled high-temperature reactors (FHRs). Potential regulatory guidance gaps identified in NUREG-2215 (NRC, 2020a) and NUREG-2216 (NRC, 2020b) relate to corrosion of non-fuel hardware, mechanical properties of cladding or coatings, and criticality safety.

For corrosion of non-fuel hardware, regulatory guidance gaps exist because new non-fuel hardware designs will evolve, and non-fuel hardware used with ARF types is not necessarily the zircaloy and stainless steel clad hardware used in light water reactors (LWRs) covered in the current regulatory guidance. Therefore, additional analyses are needed to address the potential for corrosion of these new materials in the transportation package or storage container environments. Also, spent TRISO fuel and non-fuel hardware from FHRs may have residual solid fluoride salt on the surfaces. Radiolysis of the solid fluoride salt generates fluorine, which can lead to corrosion of transportation or storage container components. The potential for a corrosive environment needs to be evaluated.

For mechanical properties of cladding and coatings, significant differences exist between LWR types and ARF types. Whereas cladding provides sole structural support for LWR fuel elements, both the cladding and the metal fuel itself may be credited for structural support of SFR fuel elements. For TRISO fuel, structural support and fission product retention do not come from cladding but from a fuel particle SiC coating layer. Also, TRISO fuel particles are formed into a cylindrical or spherical graphite matrix. Therefore, the SiC fuel coating layer and graphite matrix mechanical properties need to be evaluated for structural integrity during transportation, loading, and storage. The graphite matrix and supporting structures within the transportation canister also need to be evaluated for structural integrity if credited for configuration control during transportation. Also, the mechanical properties can change over time due to age-related degradation phenomena during extended storage. In addition to structural analyses, material property databases may be needed to characterize the metal fuel and cladding and TRISO fuel and coating layers and account for age-related degradation.

For criticality safety, the current regulatory guidance is based on LWR technology using UO_2 low enriched uranium (LEU) fuel with up to 5 percent ^{235}U and operating to an assembly-average burnup not exceeding 60 GWd/MTU. However, many advanced reactor designs will use high-assay low-enriched uranium (HALEU) fuel and operate to a burnup more than twice that of LWRs. In addition, some advanced reactors will use metal fuel that may be composed of uranium alloyed with zirconium or TRISO fuel composed of UCO. The current review methods cannot be extended to these different fuel compositions and higher burnups. Criticality evaluation codes may need to be validated for ARF. Also, radiochemical assay analyses and benchmark criticality experiments may be needed to support code validation.

Overall, the guidance in NUREG-2215 (NRC, 2020a) and NUREG-2216 (NRC, 2020b) is considered to be applicable to ARF types with only a few potential regulatory guidance gaps identified in this report. Gaps occur because current review methods provide guidance associated with LWR UO_2 fuel enriched up to 5.0 weight percent ^{235}U; however, many advanced reactor designs will use HALEU, which has a higher initial enrichment and can achieve a higher burnup. In addition, ARF does not have the same physical characteristics of traditional LWR fuel. For example, TRISO fuel can be in the form of spherical graphite matrix coated fuel particles where structural support and fission product retention is provided by the particle's SiC coating layer. This form differs from LWR fuel where UO_2 fuel pellets are contained within metal cladding for structural support and fission product retention. The differences between ARF and LWR fuel create many areas where information and technical analyses are needed to address the guidance in NUREG-2215 (NRC, 2020a) and NUREG-2216 (NRC, 2020b). Appendices I through III also identify several instances where additional information and analyses are needed to address technical questions, without a gap in the corresponding guidance (i.e., the current guidance already calls for those additional analyses to be performed). It is expected that as advanced reactor designs and transportation package/ storage cask designs progress, much of this additional information will become available.

5 REFERENCES

Alexander, D.J. and R.K. Nanstand. "The Effects of Aging for 50,000 Hours at 343°C on the Mechanical Properties of Type 308 Stainless Steel Weldments." *Proceedings of the Seventh International Symposium on Environmental Degradation of Materials in Nuclear Power Systems–Water Reactors.* Breckenridge, Colorado. NACE. Houston, Texas. pp. 747–758. 1995.

Brown, N., M. Caro, and E. Rodriguez. "Gap Analysis of Material Properties Data for Ferritic/Martensitic HT-9 Steel." GNNA-12-019. Los Alamos National Laboratory. 2012. <https://permalink.lanl.gov/object/tr?what=info:lanl-repo/lareport/LA-UR-12-24358> (Accessed date 2 August 2021).

Carmack, W., D. Porter, Y.H.S. Chang, M. Meyer, D. Burkes, C. Lee, T. Mizuno, F. Delage, and J. Somers. "Metallic Fuels for Advanced Reactors." *Journal of Nuclear Materials*. Vol. 392. pp. 139–150. 2009. <https://www.sciencedirect.com/science/article/pii/S0022311509004103> (Accessed date 24 June 2021).

Centrus. "High-Assay Low-Enriched Uranium. 2021. <https://www.centrusenergy.com/what-we-do/nuclear-fuel/high-assay-low-enriched-uranium/> (Accessed date 8 July 2021).

Chandra, K., K. Vivekanand, V.S. Raja, R. Tewari, and G.K. Dey. "Low Temperature Thermal Ageing Embrittlement of Austenitic Stainless Steel Welds and its Electrochemical Assessment." *Corrosion Science*. Vol. 54. pp. 278–290. 2012.

Code of Federal Regulations. "Packaging and Transportation of Radioactive Material." Title 10–Energy. Part 71. Washington, DC: U.S. Government Printing Office.

_____. "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste." Title 10–Energy. Part 72. Washington, DC: U.S. Government Printing Office.

Crawford, D.C., D.L. Porter, and S.L. Hayes. "Fuels for Sodium-Cooled Fast Reactors: US perspective." *Journal of Nuclear Materials*. Vol. 371. pp. 202–231. 2007.

Del Cul, G.D., B.B. Spencer, C.W. Forsberg, E.D. Collins, and W.S. Rickman. "TRISO-Coated Fuel Processing to Support High-Temperature Gas-Cooled Reactors 6.6." ORNL/TM-2002/156. September 2002. <https://info.ornl.gov/sites/publications/Files/Pub57144.pdf> (Accessed date 8 January 2019).

Demkowicz, P.A., B. Liu, and J.D. Hunn. "Coated Particle Fuel: Historical Perspectives and Current Progress." *Journal of Nuclear Materials*. Vol. 515. pp. 434–450. 2018.

DOE. "Safety Analysis Report for Fort St. Vrain Independent Spent Fuel Storage Installation." Revision 8. Chapter 3. ML103640368. Idaho Falls, Idaho: U.S. Department of Energy Idaho Operations Office. 2010.

_____. "Characteristics of Potential Repository Wastes." DOE/RW-0184-R1. Vol 2. ORNL/-2217. Vol 2. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1992.

Eidelpes, E., B.M. Hom, R.A. Hall, E. Adkins, and J. J. Jarrell. “A High-Assay Low-Enriched Uranium Fuel Transportation Concept.” *Nuclear Science and Engineering.* Vol 195:3. pp. 279–299. DOI: 10.1080/00295639.2020.1802161. 2021.

Fast Reactor Working Group (FRWG). “Nuclear Metal Fuel: Characteristics, Design, Manufacturing, Testing, and Operating History.” White Paper 18-01. ML18165A249. 2018.

Forsberg, C. and P.F. Peterson. “Spent Nuclear Fuel and Graphite Management for Salt-Cooled Reactors: Storage, Safeguards, and Repository Disposal.” *Nuclear Technology.* Vol. 191. pp. 113–121. 2015.

Garner, F.A. “Irradiation Performance of Cladding and Structural Steels in Liquid Metal Reactors.” Nuclear Materials: Part 1. Materials Science and Technology: A Comprehensive Treatment. B.R.T. Frost, editor. VCH Publishers. pp. 419–543. 1993.

Geelhood, K.J., C.E. Beyer, and W.G. Luscher. “PNNL Stress/Strain Correlation for Zircaloy,” Pacific Northwest National Laboratory. PNNL-17700. July 2008. <https://www.pnnl.gov/main/publications/external/technical_reports/PNNL-17700.pdf> (Accessed date 29 June 2021).

Gougar, H. “Baseline Concept Description of a Small Modular High Temperature Reactor,” Idaho National Laboratory. INL/EXT-14-31541, Revision 2. October 2014. <https://inldigitallibrary.inl.gov/sites/sti/sti/6531061.pdf> (Accessed date 26 July 2021).

Guenther, R.J., A.B. Johnson, A.L. Lund, E.R. Gilbert, S.P. Pednekar, F.M. Berting, L.L. Burger, S.A. Bryan, and T.M. Orlando. “Initial Evaluation of Dry Storage Issues for Spent Nuclear Fuels in Wet Storage at the Idaho Chemical Processing Plant.” INEL-96/0140. Idaho National Engineering Laboratory. 1996.

Guo, H. “Innovative Sodium Fast Reactors Control Rod Design.” GIF Symposium–Paris, France. October 2018.

Guo, H., L. Buiron, P. Sciora, and T. Kooyman. “Optimization of Reactivity Control in a Small Modular Sodium-Cooled Fast Reactor.” *Nuclear Engineering and Technology*. Vol. 52. pp. 1,367–1,379. 2020.

Hackett, M.J. and G. Povirk. “HT9 Development for the Traveling Wave Reactor, Invited.” TerraPower, LLC. *Transactions of the American Nuclear Society.* Vol. 106. Chicago, Illinois, June 24-28. 2012. <https://www.terrapower.com/wp-content/uploads/2020/01/HT9_Development_for_the_Traveling_Wave_Reactor.pdf> (Accessed date 15 June 2021).

IAEA. “Advances in High Temperature Gas Cooled Reactor Fuel Technology.” IAEA-TECDOC-1674. Vienna, Austria: International Atomic Energy Agency. 2012a. <https://www-pub.iaea.org/MTCD/Publications/PDF/TE_1674_CD_web.pdf> (Accessed date 22 June 2021).

_____. “Structural Materials for Liquid Metal Cooled Fast Reactor Fuel Assemblies: Operational Behaviour.” Nuclear Energy Series No. NF-T-4.3. Vienna, Austria: International Atomic Energy Agency. 2012b. <https://www-pub.iaea.org/MTCD/Publications/PDF/Pub1548_web.pdf> (Accessed date 24 June 2021).

_____. “High Temperature Gas Cooled Reactor Fuels and Materials.” IAEA-TECDOC-1645. Vienna, Austria: International Atomic Energy Agency. 2010. <https://www-pub.iaea.org/MTCD/Publications/PDF/TE_1645_CD/PDF/TECDOC_1645.pdf> (Accessed date 23 June 2021).

Jarrell, J. “A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium.” INL/EXT-18-51518. Idaho Falls, Idaho: Idaho National Laboratory. 2018.

Kairos Power (KP). “Technology – How it Works.” 2021. <https://kairospower.com/technology/> (Accessed date 27 July 2021).

Makenas, B.J. “Performance of Titanium Stabilized D9 Cladding and Ducts.” Westinghouse Hanford Company. HEDL-SA-3452. 1986. <https://www.osti.gov/servlets/purl/6555724> (Accessed date 17 June 2021).

Maloy, S.A. “Core Materials Development for the Fuel Cycle R&D Program.” *Journal of Nuclear Materials.* Vol. 415, Issue 3. pp. 302–305. August 2011. <https://www.sciencedirect.com/science/article/pii/S0022311511003862> (Accessed date 15 June 2021).

Maloy, S.A. “The Effects of Fast Reactor Irradiation Conditions on the Tensile Properties of Two Ferritic/Martensitic Steels.” *Journal of Nuclear Materials.* Vol. 356, Issues 1-3. pp. 62–69. September 2006.

Marsden, B. “Nuclear Graphite for High Temperature Reactors.” AEA Technology. 2001. <https://www.nrc.gov/docs/ML0117/ML011770379.pdf> (Accessed date 7 July 2021).

Marschman, S.C., F.M. Berting, R.G. Clemmer, E.R. Gilbert, R.J. Guenther, W.C. Morgan, and P. Sliva. “Characterization Plan for Fort St. Vrain and Peach Bottom Graphite Fuels.” PNNL-11365. Richland, Washington: Pacific Northwest National Laboratory. 1993.

NEA. “Technology Roadmap Update for Generation IV Nuclear Energy Systems.” OECD Nuclear Energy Agency. 2014.

NEI. “NEI Urges Revamp of Fuel Industry to Support Advanced Reactors.” 2018a. <https://www.nei.org/news/2018/revamp-of-fuel-industry-support-advanced-reactors> (Accessed date 9 July 2021).

_____. “Addressing the Challenges with Establishing the Infrastructure for the Front-End of the Fuel Cycle for Advanced Reactors.” NEI Whitepaper. 2018b. <https://www.nei.org/CorporateSite/media/filefolder/resources/reports-and-briefs/white-paper-advanced-fuel-cycle-infrastructure-201801.pdf> (Accessed date 9 July 2021).

NRC. NUREG–2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities.” Washington, DC: U.S. Nuclear Regulatory Commission. 2020a. <https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2215/index.html> (Accessed date 2 August 2021).

_____. NUREG–2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material.” Washington, DC: U.S. Nuclear Regulatory Commission. 2020b.

<https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2216/index.html> (Accessed date 2 August 2021).

Oklo Inc. "Part II. Final Safety Analysis Report." ADAMS Accession Number ML20075A003. Sunnyvale, California: Oklo Inc. 2020.

Purdue University. "Fluorine Safety." Department of Chemistry. 2021. https://www.chem.purdue.edu/chemsafety/chem/FluorineSafety.html (Accessed date 12 July 2021).

Qualls, A., N. Brown, B. Betzler, J. Carbajo, M. Greewood, R. Hale, T. Harrision, J. Powers, K. Robb, J. Terrell, and A. Wysocki. "Fluoride Salt-Cooled High-Temperature Demonstration Reactor Point Design." ORNL/TM-2016/85. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 2016.

Raj, B., D. Ramachandran, and M. Vijayalakshmi. "Development of Cladding Materials for Sodium-Cooled Fast Reactors in India." Transactions of the Indian Institute of Metals. Vol. 62, Issue 2. pp. 89–94 . April 2009. <https://link.springer.com/content/pdf/10.1007/s12666-009-0012-2.pdf> (Accessed date 17 June 2021).

Sofu, T. "Fast Reactor Fuels." ML19149A371. Washington, DC: Argonne National Laboratory. 2019. <https://www.nrc.gov/docs/ML1914/ML19149A371.pdf> (Accessed date 14 July 2021).

World Nuclear News (WNN). "TerraPower and Centrus Team Up for HALEU Production." September 16, 2020. <https://world-nuclear-news.org/Articles/TerraPower-and-Centrus-team-up-for-HALEU-productio> (Accessed date 9 July 2021).

X-energy. "Reactor: Xe-100." 2021. <https://x-energy.com/reactors/xe-100> (Accessed date 27 July 2021).

APPENDIX I

TRANSPORTATION OF FRESH FUEL

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
2.4.1 Description of Structural Design	General	Drawings and descriptive information including weights and centers of gravity	Information available on many NRC-certified package designs (NRC, 2013); Detailed description of structural design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Identification of codes and standards for package design	Codes and standards used for the package design and fabrication	Codes and standards are available (NRC, 2013), but not specifically for fuel with nonsymmetrical contents (Oklo Inc., 2020)	Applicability of codes and standards for structural design of fresh metal fuel need to be evaluated	None identified. The review method calls for verification of the code or standard developed for structures of similar design. Codes and standards to be used are expected to be defined or developed, or the technical basis be provided on the adequacy of alternative codes and standards.
2.4.2 General Requirements for All Packages	Minimum package size	The smallest overall dimension of the package must not be less than 10 cm [4 in]	Specific package dimensions are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Tamper-indicating feature	The package closure system must incorporate a tamper-indicating feature	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Positive closure	The package closure system must include a positive fastening device	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Package valve	A package valve or other device must be protected against unauthorized operation	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.3 Lifting and Tie-Down Standards for All Packages	Lifting devices	Lifting devices must be designed in accordance with 10 CFR 71.45(a)	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tie-down devices	Tie-down devices must be designed in accordance with 10 CFR 71.45(b)	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
2.4.4 General Considerations for Structural Evaluation of Packaging	Evaluation by analysis	Elements of the analysis used for structural evaluation	Specific analyses used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation by test	Elements of the test used for structural evaluation	Specific tests used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.5 Normal Conditions of Transport	Heat	Maximum temperature, maximum pressure, and thermal stress under the heat-loading condition	Structural performance of the package under the heat-loading condition is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Cold	Maximum temperature, minimum internal pressure, and residual stress under the cold condition	Structural performance of the package under the cold condition is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Reduced external pressure	Effects of reduced external pressure on the internal and external pressures of the package and the containment system	Effects of reduced external pressure is expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 2)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Increased external pressure	Effects of increased external pressure on the internal and external pressures of the package and the containment system	Effects of increased external pressure is expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Vibration and fatigue	Effects of vibration normally incident to transport; Fatigue under the combined stresses from vibration, temperature, and pressure loads	Information available on many NRC-certified package designs (NRC, 2013); however, no vibration test and analysis of packages for transporting sodium-containing fresh metal fuel are available	Sodium creep and location shift susceptibility and its effects on the geometric configuration of fresh metal fuel under the influence of vibration is to be evaluated	None identified. The review method calls for evaluating the package design for the effects of vibration. The existing review method is sufficient to call for the assessment of sodium creep under the influence of vibration.
	Water spray	Effects of the water spray test	Detailed water spray test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Free drop	Effects of the 0.3 to 1.2-m free-drop test	Information available on many NRC-certified package designs (NRC, 2013); however, no free-drop testing of packages for transporting sodium-containing fresh metal fuel is available	Sodium creep and location shift susceptibility and its effects on the geometric configuration of fresh nuclear metal fuel under the influence of drop is to be evaluated	None identified. The review method calls for evaluating the package design for the effects of free drop. The existing review method is sufficient to call for the assessment of sodium creep under the influence of drop.
	Corner drop	Not applicable because of the package weight exceedance	Not applicable	Not applicable	Not applicable
	Compression	Not applicable because of the package weight exceedance	Not applicable	Not applicable	Not applicable
	Penetration	Effects of the penetration test	Detailed penetration test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
2.4.6 Hypothetical Accident Conditions	Free drop	Effects of the 9-m free-drop test	Information available on many NRC-certified package designs (NRC, 2013); however, no 9-m free-drop testing of packages for fresh metal fuel transportation is available	Ability of the metal fuel pins to withstand the specified drop conditions and maintain containment and criticality functions is to be evaluated	None identified. The review method calls for evaluating the package design for the effects of free drop. The existing review method is sufficient to call for the assessment of metal fuel integrity under the influence of drop.
	Crush	Not applicable because of the package weight exceedance	Not applicable	Not applicable	Not applicable
	Puncture	Effects of the puncture test	Detailed puncture test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Thermal	Effects of the fire test	Detailed fire test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Immersion	Effects of the immersion test	For some package designs such as the Westinghouse Traveller package (Westinghouse Electric Company LLC, 2019), water may fill the package during the immersion test, thus applying hydrostatic pressure on the fuel rod and potentially compromising the cladding integrity	Ability of the cladding to withstand the increased external pressure from immersion test and its effects on fuel properties is to be evaluated	None identified. The review method calls for adequately evaluating the package design subjected to water pressure from immersion test. The existing review method is sufficient to call for the assessment of metal fuel integrity under the influence of water pressure from immersion test.
2.4.7 Air Transport Accident Conditions for Fissile Material		Not applicable because air transport is not anticipated	Not applicable	Not applicable	Not applicable
2.4.8 Special Requirement for Type B Packages Containing More Than 10^5 A_2		Not expected to be applicable to packages for fresh metal fuel transportation (A specific inventory estimate for the expected commercial design is needed to assess the	Not applicable	Not applicable	Not applicable

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel				
Areas of review (NUREG-2216, Chapter 2)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	applicability of 10 CFR 71.61)			
2.4.9 Air Transport of Plutonium	Not applicable because air transport of plutonium is not anticipated	Not applicable	Not applicable	Not applicable

NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Oklo Inc. "Part II. Final Safety Analysis Report." U.S. Nuclear Regulatory Commission ADAMS Accession Number ML20075A003. Sunnyvale, California: Oklo Inc. 2020.

Westinghouse Electric Company LLC. "Application for Certificate of Compliance for the Traveller PWR Fuel Shipping Package." Safety Analysis Report, Revision 1. ML19308C710. Westinghouse Electric Company LLC. 2019.

Table I-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
3.4.1 Description of the Thermal Design	Packaging design features	Drawings and description of the thermal features	Information available on many NRC-certified package designs (NRC, 2013); detailed description of thermal features is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used for the thermal design and evaluation of the package	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Content heat load specification	Maximum decay heat load; Methods and codes used to determine content decay heat loads	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Summary tables of temperatures	Maximum, minimum, and allowable temperatures of package components for normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Summary tables of pressures in the containment system	Design pressure limits of package components for normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.2 Material Properties and Component Specifications	Material thermal properties	Thermal properties of package materials; Sources of the thermal properties; Temperature-dependent thermal properties	Thermal properties for commonly used packaging materials are available; Some thermal properties of fuel pin components, structural components, and metal fuel are available (Leibowitz et al., 1976; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Accurate data to characterize phases, phase diagrams, heat capacity, and thermal properties of metal fuel are limited	None identified. The review method calls for verification of the thermal and thermomechanical properties as well as their temperature dependence. The existing review method is sufficient to deal with the assessment of required metal fuel properties important to the thermal analysis.
	Specifications of components	Maximum allowable service temperatures or pressures of package components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Thermal design limits of package materials and components	Maximum allowable temperatures of package components; Temperature limits of fuel and clad materials	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.3 General Considerations for Thermal Evaluations	Evaluation by analyses	Elements of the analysis used for thermal evaluation	Specific analyses used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation by Tests	Elements of the test used for thermal evaluation	Specific tests used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analysis	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Effects of uncertainties	Uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and analytical methods	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Conservatisms	Conservatisms associated with the thermal models and their effects on the safety parameters	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.4 Evaluation of Accessible Surface Temperatures		Thermal model used for calculating the accessible surface temperatures	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.5 Thermal Evaluation Under Normal Conditions of Transport	Heat and cold	Maximum accessible surface temperatures; Maximum temperatures of package components under the heat condition; Minimum temperatures of package components under the cold condition	The influence of heat and cold could lead to differential thermal expansion and stresses for the fuel pin components, thus potentially compromise the bonding between sodium and cladding and sodium and metal fuel slug	Thermal performance of the bonding between sodium and cladding and sodium and metal fuel slug under the heat and cold conditions is to be evaluated	None identified. The review method calls for examining that the tests for normal conditions of transport do not result in significant reduction in packaging effectiveness. The existing review method is sufficient to evaluate the performance of structure bonding and the metal fuel under the heat and cold conditions.
	Maximum normal	The maximum normal operating	Detailed thermal evaluation is expected	None expected	None identified; General acceptance criteria not impacted

Table I-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	operating pressure	pressure under the heat condition	to be provided by the applicant		by specifics of technologies
3.4.6 Thermal Evaluation Under Hypothetical Accident Conditions	Initial conditions	Initial conditions of the package	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fire test	Effects of the fire test	For the hypothetical accident conditions, the temperature of the sodium inside the fuel pin may exceed the melting point and the resulting thermal stress may compromise cladding integrity	Ability of the metal fuel pins and the cladding to withstand the hypothetical accident conditions is to be evaluated	None identified. The review method calls for examining the evaluation of the package design regarding potential consequences of the fire test. The existing review method is sufficient to evaluate the integrity of metal fuel and cladding under the conditions consistent with the fire test.
	Maximum temperatures and pressures	The maximum temperatures and pressures of the package components under the hypothetical accident conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh metal fuel				
Areas of review (NUREG-2216, Chapter 3)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
Janney, D.E. and S.L. Hayes. "Experimentally Known Properties of U-10Zr alloys: A critical Review." Nuclear Technology. Vol. 203. pp.109–128. 2018. Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 1: Phases and Phase Diagrams." Nuclear Technology. Vol. 205. pp.1,387–1,415. 2019. Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 2: Thermal and Mechanical Properties." Nuclear Technology. Vol. 206. pp.1–22. 2020. Janney, D.E. "Metallic Fuels Handbook, Part 1: Alloys Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr, Including Those with Minor Actinides (Np, Am, Cm), Rare-earth Elements (La, Ce, Pr, Nd, Gd), and Y." INL/EXT-15-36520 Revision 3 Part 1. Idaho Falls, Idaho: Idaho National Laboratory. 2018a. Janney, D.E. "Metallic Fuels Handbook, Part 2: Elements and Alloys not Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr." INL/EXT-15-36520 Revision 3 Part 2. Idaho Falls, Idaho: Idaho National Laboratory. 2018b. Leibowitz, L., E.C. Chang, M.G. Chasanov, R.L. Gibby, C. Kim, A.C. Millunzi, D. Stahl. "Properties for Liquid Metal Fast Breeder Reactor Safety Analysis." Argonne National Laboratory. ANL-CEN-RSD-76-1. 1976. NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.				

Table I-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
4.4.1 Description of the Containment System	Containment boundary	Containment design features including description of the containment boundary, containment boundary penetrations, method of closure, and leak test for penetrations.	Configuration of containment boundary varies depending on the package design for the specific contents (Westinghouse Electric Company LLC., 2019; Global Nuclear Fuel, 2018; Century Industries, 2010); Detailed description of containment design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used for the containment design of the package	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Special requirements for damaged spent nuclear fuel	Not applicable to fresh metal fuel	Not applicable	Not applicable	Not applicable

Table I-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 4)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
4.4.2 General Considerations for Containment Evaluations	Type AF fissile packages	Contents and requirement for Type AF packages	Fresh metal fuel made from uranium ore with no prior history of irradiation would presumably fall under the heading of Type AF fissile transportation packages	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Type B packages	Contents and requirement for Type B packages	Fresh metal fuel fabricated with high-assay low-enriched uranium and reprocessed uranium may have content activity that results in a Type B designation (Eidelpes et al., 2019)	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Combustible-gas Generation	Combustible gases generated in the package do not exceed 5 percent by volume	Sodium reacts violently with water, which produces sodium hydroxide and hydrogen, and the hydrogen burns when in contact with air	Measures to ensure no failure of containment boundary that would lead to violent reaction of sodium with inleakage of water producing combustible gas are to be established	None identified. The review method calls for a 5 percent concentration threshold or lower if warranted by the flammable gas. In addition, Section 7.4.10.1 requires measures to remove moisture or oxygen from the container if metallic contents could potentially support pyrophoricity. The existing review method is sufficient to evaluate the effects of reactions of sodium and fuel with water and air in the context of transport of fresh metal fuel.
4.4.3 Containment Evaluation under Normal Conditions of Transport	Type B transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for	Information available on many NRC-certified package designs (NRC, 2013; Westinghouse Electric Company LLC., 2019) and previous experience in leak testing of metal fuel pin (Burkes et al., 2009); Detailed containment	Release rate calculations and criteria used to verify cladding welds and fuel pins integrity to ensure sufficient containment by the transportation package of the sodium-containing	None identified. The review method calls for ensuring the applicant calculated the maximum permissible release rate and maximum permissible leakage rate in accordance with ANSI N14.5. The existing review method is

Table I-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		normal conditions of transport in accordance with ANSI N14.5 (ANSI, 2014)	evaluation is expected to be provided by the applicant	fuel under normal conditions of transport are to be evaluated	sufficient to evaluate metal fuel integrity under normal conditions of transport.
	Spent nuclear fuel transportation packages	Not applicable to fresh metal fuel	Not applicable	Not applicable	Not applicable
	Compliance with containment design criteria	Packages must be designed to satisfy the containment requirements of 10 CFR 71.51(a)(1) under normal conditions of transport	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.4.4 Containment Evaluation Under Hypothetical Accident Conditions	Type B transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for hypothetical accident conditions in	Information available on many NRC-certified package designs (NRC, 2013; Westinghouse Electric Company LLC., 2019) and previous experience to demonstrate containment function during hypothetical accident conditions such as the RAJ-II package (NRC, 2004; Global Nuclear Fuel,	Release rate calculations of metal fuel cladding and welds from drop, fire, and other accident conditions to ensure sufficient containment by the transportation package of the sodium-containing fuel are to be evaluated	None identified. The review method calls for no escape of krypton-85 and other radioactive material as well as no external radiation dose rate specified in 10 CFR 71.51(a)(2) for hypothetical accident conditions. The existing review method is sufficient to evaluate metal fuel integrity under

Table I-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		accordance with ANSI N14.5	2018); however, no data are available specifically for the metal fuel pin; Detailed containment evaluation is expected to be provided by the applicant		hypothetical accident conditions.
	Spent nuclear fuel transportation packages	Not applicable to fresh metal fuel	Not applicable	Not applicable	Not applicable
	Compliance with containment design criteria	Packages must be designed to satisfy the containment requirements of 10 CFR 71.51(a)(2) under hypothetical accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh metal fuel				
Areas of review (NUREG-2216, Chapter 4)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**

ANSI. ANSI N14.5-2014, “American National Standard for Radioactive Materials–Leakage Tests on Packages for Shipment.” New York, New York: American National Standards Institute. 2014.

Burkes, D., R. Fielding, D. Porter, D. Crawford, and M. Meyer. “A US Perspective on Fast Reactor Fuel Fabrication Technology and Experience Part I: Metal Fuels and Assembly Design.” Journal of Nuclear Materials. Vol. 389. Pp. 458–469. 2009.

Century Industries. “Safety Analysis Report for the Century Industries Versa-Pac Shipping Container.” Revision 3. ML101110135. Bristol, Virginia: Century Industries. 2010.

Eidelpes, E., J.J. Jarrell, H.E. Adkins, B.M. Hom, J.M. Scaglione, R.A. Hall, and B.D. Brickner. “UO2 HALEU Transportation Package Evaluation and Recommendations.” INL/EXT-19-56333. Idaho Falls, Idaho: Idaho National Laboratory. 2019.

Global Nuclear Fuel. “RAJ-II Safety Analysis Report.” Revision 10. ML18247A218. Wilmington, North Carolina: Global Nuclear Fuel-Americas, LLC. 2018.

NRC. Safety Evaluation Report for Certificate of Compliance No. 9309 Model No. RAJ-II Package Revision No. 0. ML043360200. Washington, DC: U.S. Nuclear Regulatory Commission. 2004.

_____. NUREG–0383, “Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance.” Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Westinghouse Electric Company LLC. “Application for Certificate of Compliance for the Traveller PWR Fuel Shipping Package. Safety Analysis Report. Revision 0.” Westinghouse Electric Company LLC. 2019.

Table I-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.4.1 Description of the Shielding Design	Shielding design features	Drawings and description of the shielding design features	Transportation packages for fresh fuel that is classified as Type AF quantity material usually do not require shielding, such as the Westinghouse Traveller package "Type A" configuration (Westinghouse Electric Company LLC, 2019) and the Versa-Pac package (NRC, 2020)	Shielding design for fresh metal fuel with diverse sources of uranium and fabricated with different reprocessing methods is to be evaluated	None identified. The review method calls for evaluating a description of the shielding design features to ensure it addresses those items important to the package's shielding performance. The existing review method is sufficient to evaluate the shielding design dealing with metal fuel fabricated with reprocessed uranium.
	Summary tables of maximum external radiation levels	Maximum radiation levels for all relevant package surfaces and appropriate distances from these surfaces under normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.2 Radioactive Materials and Source Terms	Source-term calculation methods	Methods used to determine the bounding source terms for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Gamma sources	Gamma source strengths and spectra for the package contents	Metal fuel fabricated with reprocessed uranium could possibly contain radioactive sources; Although ASTM C996 (ASTM, 2021) is frequently applied, information in the literature is limited	Source-term specification including gamma sources and their energies arising from fission products from reprocessed uranium or other impurities is to be evaluated	None identified. The review method calls for ensuring the application provides activity (or mass) and total inventory of radionuclides that contribute significantly to the source term. The existing review method is sufficient to evaluate residual gamma sources in metal fuel fabricated with reprocessed uranium.
	Neutron sources	Neutron source strengths and spectra for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 5)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
5.4.3 Shielding Model and Model Specifications	Configuration of source and shielding	Dimensions and materials properties of the package contents, radioactive sources in the contents, and the packaging components	Information available on many NRC-certified package designs (NRC, 2013) and the Westinghouse Traveller package "Type B" configuration (Westinghouse Electric Company LLC, 2019); Detailed shielding evaluation in a Type BF configuration is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Material properties (e.g., composition, mass densities, and atom densities) of packaging components, package contents, and the conveyance	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.4 Shielding Evaluation	Methods	Methods used for the shielding evaluations under normal and accident conditions	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 5)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Code input and output data	Key input data and output files for the shielding evaluations	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fluence-rate-to-radiation-level conversion factors	Accuracy and acceptance of the conversion factors	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	External radiation levels	External radiation levels under normal and accident conditions	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

ASTM. C996. “Standard Specification for Uranium Hexafluoride Enriched to Less Than 5% ^{235}U” <https://compass.astm.org/EDIT/html_annot.cgi?C996+20> (Accessed May 27, 2021). 2021.

NRC. NUREG–0383, “Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance.” Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

_____. “Certificate of Compliance for Radioactive Material Packages. Certificate No. 9342.” Revision 15. ML20139A034. Washington, DC: U.S. Nuclear Regulatory Commission. <https://rampac.energy.gov/docs/default-source/certificates/1019342.pdf>.(Accessed May 27, 2021). 2020.

Westinghouse Electric Company LLC. “Application for Certificate of Compliance for the Traveller PWR Fuel Shipping Package.” Safety Analysis Report. Revision 1. ML19308C710. Westinghouse Electric Company LLC. 2019.

Table I-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
6.4.1 Description of Criticality Design	Packaging design features	Design features important for criticality safety	Information available on many NRC-certified package designs (NRC, 2013); Detailed description of criticality features is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used in all aspects of the criticality design and evaluation	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Summary table of criticality evaluations	Maximum value of k_{eff}, uncertainty, bias and bias uncertainty for all relevant cases; Number of packages evaluated in the array cases	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Criticality safety index (CSI)	CSI limits for all package configurations	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.2 Fissile material contents		Content and type of fissile material	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted

Table I-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
					by specifics of technologies
6.4.3 General Considerations for Criticality Evaluations	Model configuration	Criticality evaluations demonstrating subcritical margins for single package and package arrays under normal and hypothetical conditions of transport	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Materials and their properties used in the criticality models	Information is expected to be available at the time of an application	None expected	None identified, because this is a reporting of materials used for the criticality evaluation
	Analysis methods and nuclear data	Computer code and cross-section library used for criticality evaluations	Detailed criticality evaluation is expected to be provided by the applicant	None expected	None identified; The computer codes and cross section libraries are not impacted by specifics of technologies
	Demonstration of maximum reactivity	Analyses demonstrate the maximum k_{eff}	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Confirmatory analyses	Confirmatory analysis of the criticality calculations	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Moderator exclusion under hypothetical accident conditions	Package subcriticality under hypothetical accident conditions	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.4 Single Package Evaluation	Configuration	Models for criticality evaluations confirming subcritical margins maintained for single package under normal and hypothetical accident conditions of transport	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Results	Results of the criticality calculations for single package	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
6.4.5 Evaluations of Package Arrays	Package arrays under normal conditions of transport	Criticality evaluation for an array of 5N packages that is subcritical under normal conditions of transport	Information available on many NRC-certified package designs (NRC, 2013). Although no prior experience for fresh metal fuel package, information is expected to be available at the time of an application.	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation of package arrays under hypothetical accident conditions	Criticality evaluation for an array of 2N packages that is subcritical under hypothetical accident conditions	Information available on many NRC-certified package designs (NRC, 2013). Although no prior experience for fresh metal fuel package, information is expected to be available at the time of an application.	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Package arrays results and criticality safety index	Appropriate N value is used to ascertain the CSI	Information available on many NRC-certified package designs (NRC, 2013). Although no prior experience for fresh metal fuel package, information is expected to be available at the time of an application.	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
6.4.6 Benchmark Evaluations	Experiments and applicability	Benchmarking computer codes for criticality calculations against fitting critical experiments	Information available on many NRC-certified package designs (NRC, 2013); however, no information for sodium-containing fresh metal fuel was found. Criticality benchmark data and validation of existing criticality codes and methods are limited for enrichments between 5 and 20 weight percent (Jarrell, 2018).	Criticality benchmark data and applicability of existing criticality codes and methods for fresh metal fuel with enrichments between 5 and 20 weight percent ^{235}U are to be evaluated	None identified. The review method calls for verifying the applicant has benchmarked the computer codes used for criticality calculations against appropriate critical experiments applicable to the actual packaging design and contents. The existing review method is sufficient to deal with the availability of criticality benchmark data and applicability of existing criticality codes and methods for fresh metal fuel.

Table I-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Bias determination	Results of the benchmark calculations and bias evaluations	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001). No criticality benchmarking for fresh metal fuel with higher enrichment was found.	Criticality benchmarking for fresh metal fuel with higher enrichment is to be evaluated, given the potential lack of criticality benchmark data	None identified. The review method calls for evaluating whether the applicant demonstrates that the benchmark calculations are adequately converged and justifies the bias and bias uncertainty. The existing review method is sufficient to deal with criticality benchmarking for fresh metal fuel.
6.4.7 Burnup Credit Evaluation for Commercial Light-Water Reactor Spent Nuclear Fuel		Not applicable because of unirradiated fuel	Not applicable	Not applicable	Not applicable

ANS. American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998 (R2007). "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." La Grange Park, Illinois: American Nuclear Society. 2007.
Jarrell, J. "A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium." INL/EXT-18-51518. Idaho Falls, Idaho: Idaho National Laboratory. 2018.
NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.
_____. NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology." Oak Ridge, Tennessee: Science Applications International Corporation. U.S. Nuclear Regulatory Commission. 2001.
_____. NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages." ORNL/TM-11936. Oak Ridge, TN: Oak Ridge National Laboratory. U.S. Nuclear Regulatory Commission. 1997.

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.4.1 Drawings		Content of drawings	Information available on many NRC-certified package designs (NRC, 2013); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for examining the content of engineering drawings, as well as the description of materials in package designs .
7.4.2 Codes and Standards	Usage and endorsement	Codes and standards used for the package design and construction	Codes and standards are available (NRC, 2013); however, it is uncertain whether those standards would apply to new materials potentially to be used for package design and fabrication for transport of metal fuel	Applicability of codes and standards for package design and fabrication with new materials is to be evaluated	None identified. The review method calls for verification of the codes and standards for packaging components important to safety. Codes and standards to be used are expected to be defined or developed, or the technical basis be provided for the adequacy of alternative codes and standards.

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	ASME code components	Construction of ASME code components	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code case use/acceptability	Acceptability of ASME code cases	Specific code case referenced is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Non-ASME code components	Construction of non-ASME code components	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.3 Weld Design and Inspection	Weld Design and Inspection	Welding criteria and weld procedure qualification requirements	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that standard welding processes are adequate for package design and fabrication for transport of metal fuel. If new technologies were used in the design and fabrication of welds, the SRP calls for examination of compliance with any established codes and standards proposed in the application on design and construction.

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Moderator exclusion for commercial spent nuclear fuel packages under hypothetical accident conditions	Not applicable to packages for fresh metal fuel transportation	Not applicable	Not applicable	Not applicable
7.4.4 Mechanical Properties	Tensile properties	Acceptability of material tensile properties	Mechanical properties for commonly used packaging materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that commonly used packaging materials may be also adequate for package design and fabrication for transport of metal fuel. If alternative or new materials were required in the design and fabrication of transportation packages, the SRP calls for examination of the adequacy of information in the

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
					application related to mechanical properties of those alternative materials.
	Fracture resistance	Acceptability of material fracture toughness	Mechanical properties for commonly used packaging materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tensile properties and creep of aluminum alloys at elevated temperatures	Acceptability of the tensile properties and creep of aluminum alloys	Mechanical properties for commonly used aluminum alloys are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Impact limiters	Acceptability of the mechanical properties of the impact limiter materials	Mechanical properties for commonly used impact limiter materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.4.5 Thermal Properties of Materials		Thermal properties of package materials; Effect of degradation and anisotropic dependencies of thermal properties	Thermal properties for commonly used packaging materials are available; Some thermal properties of fuel pin components, structural components, and metal fuel are available (Leibowitz et al., 1976; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Accurate data to characterize phases, phase diagrams, heat capacity, and thermal properties of metal fuel are limited	None identified. The review method calls for verification of the thermal properties and the change in these properties from material degradation. The existing review method is sufficient to deal with the availability of information related to metal fuel properties important to the thermal analysis.
7.4.6 Radiation Shielding	Neutron-shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Gamma-shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.7 Criticality Control	Neutron-absorbing (poison) material specification	Chemical composition, physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; Qualification testing	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Computation of percent credit for boron-based neutron absorbers	Level of credit allowed for absorber materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Qualifying properties not	Qualification of absorber material properties not	Detailed evaluation of packaging materials is	None expected	None identified; General acceptance criteria not impacted

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	associated with attenuation	associated with neutron attenuation	expected to be provided by the applicant		by specifics of technologies
7.4.8 Corrosion Resistance	Environments	Range of environmental conditions encountered for package components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Carbon and low-alloy steels	Environment dependencies of corrosion rate; Coatings for corrosion prevention	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Austenitic stainless steel	Localized corrosion and chloride-induced stress corrosion cracking in chloride-containing environments; Intergranular corrosion and stress corrosion cracking in sensitized stainless steel	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.9 Protective Coatings	Review guidance	Coating specifications	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
					by specifics of technologies
	Scope of coating application	Purpose of the coating, lists the components to be coated, and the expected environmental conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating selection	Coating manufacturer, type of primers and topcoat, coating thickness, and ability of the coating to withstand the in-service conditions	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating qualification testing	Qualification testing for coating performance in accordance with several standard ASTM (and possibly other) tests	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for evaluating any qualification testing for the demonstration of coating performance.

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.10 Content Reactions	Flammable and explosive reactions	Effects of flammable and explosive reactions among the content materials	Sodium reacts violently with water, which produces sodium hydroxide and hydrogen, and the hydrogen burns when in contact with air	Safety protocols in transporting sodium-containing metal fuel are to be established	None identified. The review method calls for measures to remove moisture or oxygen to be demonstrated. The existing review method is sufficient to evaluate the effects of reactions of sodium and fuel with water in the context of transport of fresh metal fuel.
	Content chemical reactions, outgassing, and corrosion	Effects of chemical reactions, outgassing, and corrosion among the contents and between the contents and the package components	Some data are available on metal fuel (Carmack et al., 2009; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Effects of air and water on chemical interaction and galvanic coupling of package internal materials including sodium in the fuel pin are to be evaluated	Additional guidance may need to be developed to address the corrosion of non-fuel hardware associated with metal fuel. The SRP calls for examining that corrosion wastage will not lead to a loss of intended functions; however, for non-fuel hardware the current review method is limited to guidance for the examination

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
					of corrosion of hardware components associated with stainless steel or zirconium alloy-clad UO_2 fuels.
7.4.11 Radiation Effects		Effects of radiation on the performance of the package materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. Commonly used packaging materials may be also adequate for package design and fabrication for transport of metal fuel. If alternative or new materials were required in the design and fabrication of transportation packages, the SRP calls for examination of the adequacy of information in the application related to radiation effects on

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel				
Areas of review (NUREG-2216, Chapter 7)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
				those alternative materials.
7.4.12 Package Contents	Chemical and physical form of the package contents; Effects of corrosion, chemical reactions, and radiation on the properties of the contents	Some data are available on metal fuel (Carmack et al., 2009; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Effects of air and water on chemical interaction and galvanic coupling of package internal materials including sodium in the fuel pin are to be evaluated	None identified. The review method calls for evaluating effects of corrosion, chemical reactions, and radiation. The existing review method is sufficient to evaluate the effects of air and water on chemical interaction and galvanic coupling of the package for transporting fresh metal fuel.

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.13 Fresh (Unirradiated) Fuel Cladding		Mechanical properties of the cladding materials; Sources of cladding materials data	Some data are available on metal fuel cladding materials (Carmack et al., 2009; Garner, 1993; IAEA, 2012)	Advanced cladding material properties that can be used to achieve high burnup are to be evaluated, especially material performance data under the influence of irradiation	Additional guidance may need to be developed to address the mechanical properties of advanced cladding materials for metal fuel. The current review method is limited to guidance for the examination of mechanical properties of zirconium and aluminum alloy cladding.
7.4.14 Spent Nuclear Fuel	Spent fuel classification	Not applicable to fresh metal fuel	Not applicable	Not applicable	Not applicable
	Uncanned spent fuel	Not applicable to fresh metal fuel	Not applicable	Not applicable	Not applicable
	Canned spent fuel	Not applicable to fresh metal fuel	Not applicable	Not applicable	Not applicable

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.15 Bolting Material		Material properties of the bolting; Effects of corrosion, chemical reactions, and radiation on the bolting materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.16 Seals	Metallic seals	Material properties of metallic seals; Effects of corrosion, chemical reactions, and radiation on the seal materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Elastomeric seals	Material properties of elastomeric seals; Effects of corrosion, chemical reactions, and radiation on the seal materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel

Areas of review (NUREG-2216, Chapter 7)	Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps

Carmack, W., D. Porter, Y.H.S. Chang, M. Meyer, D. Burkes, C. Lee, T. Mizuno, F. Delage, and J. Somers. "Metallic Fuels for Advanced Reactors." *Journal of Nuclear Materials.* Vol. 392. pp. 139–150. 2009.

Garner, F.A. "Irradiation Performance of Cladding and Structural Steels in Liquid Metal Reactors." Nuclear Materials: Part 1. Materials Science and Technology: A Comprehensive Treatment. Frost, B.R.T., Editor. VCH Publishers. pp. 419–543. 1993.

IAEA. "Structural Materials for Liquid Metal Cooled Fast Reactor Fuel Assemblies: Operational Behaviour." Nuclear Energy Series No. NF-T-4.3. Vienna, Austria: International Atomic Energy Agency. 2012.

Janney, D.E. "Metallic Fuels Handbook, Part 1: Alloys Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr, Including Those with Minor Actinides (Np, Am, Cm), Rare-earth Elements (La, Ce, Pr, Nd, Gd), and Y." INL/EXT-15-36520 Revision 3 Part 1. Idaho Falls, Idaho: Idaho National Laboratory. 2018a.

Janney, D.E. "Metallic Fuels Handbook, Part 2: Elements and Alloys not Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr." INL/EXT-15-36520 Revision 3 Part 2. Idaho Falls, Idaho: Idaho National Laboratory. 2018b.

Janney, D.E. and S.L. Hayes. "Experimentally Known Properties of U-10Zr alloys: A critical Review." *Nuclear Technology*. Vol. 203. pp.109–128. 2018.

Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 1: Phases and Phase Diagrams." *Nuclear Technology*. Vol. 205. pp.1,387–1,415. 2019.

Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 2: Thermal and Mechanical Properties." *Nuclear Technology*. Vol. 206. pp.1–22. 2020.

NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table I-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel

Areas of review (NUREG-2216, Chapter 2)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
2.4.1 Description of Structural Design	General	Drawings and descriptive information including weights and centers of gravity	Information available on many NRC-certified package designs, particularly the Versa-Pac package for transport of fresh TRISO fuel (NRC, 2013; Century Industries, 2009, 2010); Detailed description of structural design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Identification of codes and standards for package design	Codes and standards used for the package design and fabrication	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.2 General Requirements for All Packages	Minimum package size	The smallest overall dimension of the package must not be less than 10 cm [4 in]	Specific package dimensions are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tamper-indicating feature	The package closure system must incorporate a tamper-indicating feature	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Positive closure	The package closure system must include a positive fastening device	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Package valve	A package valve or other device must be protected against unauthorized operation	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.3 Lifting and Tie-Down Standards for All Packages	Lifting devices	Lifting devices must be designed in accordance with 10 CFR 71.45(a)	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tie-down devices	Tie-down devices must be designed in accordance with 10 CFR 71.45(b)	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.4 General Considerations for Structural Evaluation of Packaging	Evaluation by analysis	Elements of the analysis used for structural evaluation	Specific analyses used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation by test	Elements of the test used for structural evaluation	Specific tests used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
2.4.5 Normal Conditions of Transport	Heat	Maximum temperature, maximum pressure, and thermal stress under the heat-loading condition	Structural performance of the package under the heat-loading condition is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Cold	Maximum temperature, minimum internal pressure, and residual stress under the cold condition	Structural performance of the package under the cold condition is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Reduced external pressure	Effects of reduced external pressure on the internal and external pressures of the package and the containment system	Effects of reduced external pressure is expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Increased external pressure	Effects of increased external pressure on the internal and external pressures of the package and the containment system	Effects of increased external pressure is expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Vibration and fatigue	Effects of vibration normally incident to transport; Fatigue under the combined stresses from vibration, temperature, and pressure loads	Effects of vibration and fatigue analysis are expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Water spray	Effects of the water spray test	Detailed water spray test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Free drop	Effects of the 0.3 to 1.2-m free-drop test	Detailed free-drop test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Corner drop	Not applicable because of the package weight exceedance	Not applicable	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Compression	Not applicable because of the package weight exceedance	Not applicable	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Penetration	Effects of the penetration test	Detailed penetration test is expected to be	None expected	None identified; General acceptance criteria not impacted

Table I-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
			provided by the applicant		by specifics of technologies
2.4.6 Hypothetical Accident Conditions	Free drop	Effects of the 9-m free-drop test	Detailed free-drop test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Crush	Not applicable because of the package weight exceedance	Not applicable	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Puncture	Effects of the puncture test	Detailed puncture test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Thermal	Effects of the fire test	Detailed fire test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Immersion	Effects of the immersion test	Detailed immersion test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.7 Air Transport Accident Conditions for Fissile Material		Not applicable because air transport is not anticipated	Not applicable	Not applicable	Not applicable

Table I-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel				
Areas of review (NUREG-2216, Chapter 2)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
2.4.8 Special Requirement for Type B Packages Containing More Than 10^5 A_2	Not expected to be applicable to packages for fresh metal fuel transportation (A specific inventory estimate for the expected commercial design is needed to assess the applicability of 10 CFR 71.61)	Not applicable	Not applicable	Not applicable
2.4.9 Air Transport of Plutonium	Not applicable because air transport of plutonium is not anticipated	Not applicable	Not applicable	Not applicable
Century Industries. "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container." Revision 0. ML092321070. Bristol, Virginia: Century Industries. 2009. Century Industries. "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container." Revision 3. ML101110135. Bristol, Virginia: Century Industries. 2010. NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.				

Table I-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
3.4.1 Description of the Thermal Design	Packaging design features	Drawings and description of the thermal features	Information available on many NRC-certified package designs, particularly the Versa-Pac package for transport of fresh TRISO fuel (NRC, 2013; Century Industries, 2009, 2010); Detailed description of thermal features is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used for the thermal design and evaluation of the package	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Content heat load specification	Maximum decay heat load; Methods and codes used to determine content decay heat loads	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Summary tables of temperatures	Maximum, minimum, and allowable temperatures of package components for normal and	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		accident conditions			
	Summary tables of pressures in the containment system	Design pressure limits of package components for normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.2 Material Properties and Component Specifications	Material thermal properties	Thermal properties of package materials; Sources of the thermal properties; Temperature-dependent thermal properties	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Specifications of components	Maximum allowable service temperatures or pressures of package components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Thermal design limits of package materials and components	Maximum allowable temperatures of package components; Temperature limits of fuel and clad materials	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
3.4.3 General Considerations for Thermal Evaluations	Evaluation by analyses	Elements of the analysis used for thermal evaluation	Specific analyses used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation by Tests	Elements of the test used for thermal evaluation	Specific tests used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analysis	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Effects of uncertainties	Uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and analytical methods	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Conservatisms	Conservatisms associated with the thermal models and their effects on the safety parameters	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.4 Evaluation of Accessible Surface Temperatures		Thermal model used for	Detailed thermal evaluation is expected	None expected	None identified; General acceptance criteria not impacted

Table I-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh TRISO fuel

Areas of review (NUREG-2216, Chapter 3)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
		calculating the accessible surface temperatures	to be provided by the applicant		by specifics of technologies
3.4.5 Thermal Evaluation Under Normal Conditions of Transport	Heat and cold	Maximum accessible surface temperatures; Maximum temperatures of package components under the heat condition; Minimum temperatures of package components under the cold condition	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Maximum normal operating pressure	The maximum normal operating pressure under the heat condition	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.6 Thermal Evaluation Under Hypothetical Accident Conditions	Initial conditions	Initial conditions of the package	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Fire test	Effects of the fire test	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Maximum temperatures and pressures	The maximum temperatures and pressures of the package components under the hypothetical accident conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
Century Industries. "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container." Revision 0. ML092321070. Bristol, Virginia: Century Industries. 2009. Century Industries. "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container." Revision 3. ML101110135. Bristol, Virginia: Century Industries. 2010. NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.					

Table I-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
4.4.1 Description of the Containment System	Containment boundary	Containment design features including description of the containment boundary, containment boundary penetrations, method of closure, and leak test for penetrations.	Information available on many NRC-certified package designs, particularly the Versa-Pac package for transport of fresh TRISO fuel (NRC, 2013; Century Industries, 2009, 2010); Detailed description of containment design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used for the containment design of the package	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Special requirements for damaged spent nuclear fuel	Not applicable to fresh TRISO fuel	Not applicable	Not applicable	Not applicable

Table I-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh TRISO fuel

Areas of review (NUREG-2216, Chapter 4)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
4.4.2 General Considerations for Containment Evaluations	Type AF fissile packages	Contents and requirement for Type AF packages	Fresh TRISO fuel made from uranium ore with no prior history of irradiation would presumably fall under the heading of Type AF fissile transportation packages	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Type B packages	Contents and requirement for Type B packages	Fresh TRISO fuel fabricated with high-essay low-enriched uranium and reprocessed uranium may have content activity that results in a Type B designation (Eidelpes et al., 2019)	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Combustible-gas Generation	Combustible gases generated in the package do not exceed 5 percent by volume	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
4.4.3 Containment Evaluation under Normal Conditions of Transport	Type B transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for normal conditions of transport in accordance with ANSI N14.5 (ANSI, 2014)	Existing NRC-certified Versa-Pac package provides all applicable containment evaluation under the normal conditions of transport; Detailed package containment evaluations under normal conditions of transport are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Spent nuclear fuel transportation packages	Not applicable to fresh TRISO fuel	Not applicable	Not applicable	Not applicable
	Compliance with containment design criteria	Packages must be designed to satisfy the containment requirements of 10 CFR 71.51(a)(1) under normal conditions of transport	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.4.4 Containment Evaluation Under	Type B transportation packages	Releasable source term, maximum permissible release rate,	Existing NRC-certified Versa-Pac package provides applicable containment evaluation	None expected	None identified; General acceptance criteria not impacted

Table I-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
Hypothetical Accident Conditions		maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for hypothetical accident conditions in accordance with ANSI N14.5	under hypothetical accident conditions; Detailed package containment evaluation under hypothetical accident conditions is expected to be provided by the applicant		by specifics of technologies
	Spent nuclear fuel transportation packages	Not applicable to fresh TRISO fuel	Not applicable	Not applicable	Not applicable
	Compliance with containment design criteria	Packages must be designed to satisfy the containment requirements of 10 CFR 71.51(a)(2) under hypothetical accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

ANSI. ANSI N14.5-2014, “American National Standard for Radioactive Materials–Leakage Tests on Packages for Shipment.” New York, New York: American National Standards Institute. 2014.
Century Industries. “Safety Analysis Report for the Century Industries Versa-Pac Shipping Container.” Revision 0. ML092321070. Bristol, Virginia: Century Industries. 2009.
Century Industries. “Safety Analysis Report for the Century Industries Versa-Pac Shipping Container.” Revision 3. ML101110135. Bristol, Virginia: Century Industries. 2010.
Eidelpes, E., J.J. Jarrell, H.E. Adkins, B.M. Hom, J.M. Scaglione, R.A. Hall, and B.D. Brickner. “UO2 HALEU Transportation Package Evaluation and Recommendations.” INL/EXT-19-56333. Idaho Falls, Idaho: Idaho National Laboratory. 2019.
NRC. NUREG–0383, “Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance.” Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table I-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.4.1 Description of the Shielding Design	Shielding design features	Drawings and description of the shielding design features	The Versa-Pac package is certified to transport fresh TRISO fuel. Since gamma and neutron shielding are not required for the contents transported in the Versa-Pac package, no shielding evaluation is performed for this package (Century Industries, 2010).	Shielding design for fresh TRISO fuel with diverse sources of reprocessed uranium is to be evaluated	None identified. The review method calls for evaluating a description of the shielding design features to ensure it addresses those items important to the package's shielding performance. The existing review method is sufficient to support the evaluation of shielding performance of TRISO fuel fabricated with reprocessed uranium.
	Summary tables of maximum external radiation levels	Maximum radiation levels for all relevant package surfaces and appropriate distances from these surfaces under normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.2 Radioactive Materials and Source Terms	Source-term calculation methods	Methods used to determine the bounding source	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted

Table I-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		terms for the package contents			by specifics of technologies
	Gamma sources	Gamma source strengths and spectra for the package contents	TRISO fuel fabricated with reprocessed uranium could possibly contain radioactive sources; Although ASTM C996 (ASTM, 2021) is frequently applied, information in the literature is limited	Source-term specification including gamma sources and their energies arising from fission products from reprocessed uranium or other impurities is to be evaluated	None identified. The review method specifies that the application provides activity and total inventory of radionuclides contributing significantly to the source term. The existing review method is sufficient to evaluate the source-term specification for TRISO fuel fabricated with reprocessed uranium.
	Neutron sources	Neutron source strengths and spectra for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.3 Shielding Model and Model Specifications	Configuration of source and shielding	Dimensions and materials properties of the package contents, radioactive sources in the contents, and the	Information available on many NRC-certified package designs (NRC, 2013); Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		packaging components			
	Material properties	Material properties (e.g., composition, mass densities, and atom densities) of packaging components, package contents, and the conveyance	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.4 Shielding Evaluation	Methods	Methods used for the shielding evaluations under normal and accident conditions	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code input and output data	Key input data and output files for the shielding evaluations	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fluence-rate-to-radiation-level conversion factors	Accuracy and acceptance of the conversion factors	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	External radiation levels	External radiation levels under normal and	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted

Table I-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		accident conditions			by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

ASTM. C996. “Standard Specification for Uranium Hexafluoride Enriched to Less Than 5% ^{235}U” <https://compass.astm.org/EDIT/html_annot.cgi?C996+20> (Accessed May 27, 2021). 2021.

Century Industries. “Safety Analysis Report for the Century Industries Versa-Pac Shipping Container.” Revision 3. ML101110135. Bristol, Virginia: Century Industries. 2010.

NRC. NUREG–0383, “Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance.” Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table I-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
6.4.1 Description of Criticality Design	Packaging design features	Design features important for criticality safety	Information available on many NRC-certified package designs, particularly the Versa-Pac package for transport of fresh TRISO fuel (NRC, 2013, 2020); Detailed description of criticality features is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used in all aspects of the criticality design and evaluation	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Summary table of criticality evaluations	Maximum value of k_{eff}, uncertainty, bias and bias uncertainty for all relevant cases; Number of packages evaluated in the array cases	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Criticality safety index (CSI)	CSI limits for all package configurations	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted

Table I-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
					by specifics of technologies
6.4.2 Fissile material contents		Content and type of fissile material	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.3 General Considerations for Criticality Evaluations	Model configuration	Criticality evaluations demonstrating subcritical margins for single package and package arrays under normal and hypothetical conditions of transport	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Materials and their properties used in the criticality models	Information is expected to be available at the time of an application	None expected	None identified, because this is a reporting of materials used for the criticality evaluation
	Analysis methods and nuclear data	Computer code and cross-section library used for criticality evaluations	Detailed criticality evaluation is expected to be provided by the applicant	None expected	None identified; The computer codes and cross section libraries are not impacted by specifics of technologies

Table I-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Demonstration of maximum reactivity	Analyses demonstrate the maximum k_{eff}	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Confirmatory analysis of the criticality calculations	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Moderator exclusion under hypothetical accident conditions	Package subcriticality under hypothetical accident conditions	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.4 Single Package Evaluation	Configuration	Models for criticality evaluations confirming subcritical margins maintained for single package under normal and hypothetical accident conditions of transport	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Results	Results of the criticality calculations for single package	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
6.4.5 Evaluations of Package Arrays	Package arrays under normal conditions of transport	Criticality evaluation for an array of 5N packages that is subcritical under normal conditions of transport	Information available on many NRC-certified package designs, particularly the Versa-Pac package for transport of fresh TRISO fuel (NRC, 2013, 2020); Detailed criticality evaluations are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation of package arrays under hypothetical accident conditions	Criticality evaluation for an array of 2N packages that is subcritical under hypothetical accident conditions	Detailed criticality evaluations are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Package arrays results and criticality safety index	Appropriate N value is used to ascertain the CSI	Detailed criticality evaluations are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
6.4.6 Benchmark Evaluations	Experiments and applicability	Benchmarking computer codes for criticality calculations against fitting critical experiments	Information available on many NRC-certified package designs, particularly the Versa-Pac package for transport of fresh TRISO fuel (NRC, 2013, 2020); however, criticality benchmark data and validation of existing criticality codes and methods are limited for enrichments between 5 and 20 weight percent (Jarrell, 2018)	Criticality benchmark data and applicability of existing criticality codes and methods for fresh TRISO fuel with enrichments between 5 and 20 weight percent ^{235}U are to be evaluated	None identified. The review method calls for verifying the applicant has benchmarked the computer codes used for criticality calculations against appropriate critical experiments applicable to the actual packaging design and contents. The existing review method is sufficient to deal with the availability of criticality benchmark data and applicability of existing criticality codes and methods for fresh TRISO fuel.

Table I-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Bias determination	Results of the benchmark calculations and bias evaluations	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001). Criticality benchmarking for fresh TRISO fuel with higher enrichment is limited.	Criticality benchmarking for fresh TRISO fuel with higher enrichment is to be evaluated, given the potential lack of criticality benchmark data	None identified. The review method calls for evaluating whether the applicant demonstrates that the benchmark calculations are adequately converged and justifies the bias and bias uncertainty. The existing review method is sufficient to deal with criticality benchmarking for fresh TRISO fuel.
6.4.7 Burnup Credit Evaluation for Commercial Light-Water Reactor Spent Nuclear Fuel		Not applicable because of unirradiated fuel	Not applicable	Not applicable	Not applicable

ANS. American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998 (R2007). "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." La Grange Park, Illinois: American Nuclear Society. 2007.

Jarrell, J. "A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium." INL/EXT-18-51518. Idaho Falls, Idaho: Idaho National Laboratory. 2018.

NRC. "Safety Evaluation Report for Model No. Versa-Pac Package Certificate of Compliance No. 9342 Revision No. 15." ML20139A037. Washington, DC: U.S. Nuclear Regulatory Commission. 2020.

_____. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

_____. NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology." Oak Ridge, Tennessee: Science Applications International Corporation. U.S. Nuclear Regulatory Commission. 2001.

_____. NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages." ORNL/TM-11936. Oak Ridge, TN: Oak Ridge National Laboratory. U.S. Nuclear Regulatory Commission. 1997.

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.4.1 Drawings		Content of drawings	Information available on many NRC-certified package designs, particularly the Versa-Pac package for transport of fresh TRISO fuel (NRC, 2013; Century Industries, 2009, 2010); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for examining the content of engineering drawings, as well as the description of materials in package designs.
7.4.2 Codes and Standards	Usage and endorsement	Codes and standards used for the package design and construction	Codes and standards are available (NRC, 2013); however, it is uncertain whether those standards would apply to new materials potentially to be used for package design and fabrication for transport of TRISO fuel	Applicability of codes and standards for package design and fabrication with new materials is to be evaluated	None identified. The review method calls for verification of the codes and standards for packaging components important to safety. Codes and standards to be used are expected to be defined, or the technical basis be provided for the adequacy of alternative codes and standards.

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	ASME code components	Construction of ASME code components	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code case use/acceptability	Acceptability of ASME code cases	Specific code case referenced is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Non-ASME code components	Construction of non-ASME code components	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.3 Weld Design and Inspection	Weld Design and Inspection	Welding criteria and weld procedure qualification requirements	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that standard welding processes are adequate for package design and fabrication for transport of TRISO fuel. If new technologies were used in the design and fabrication of welds, the SRP calls for examination of compliance with any established codes and standards proposed in the application on design and construction.
	Moderator exclusion for commercial spent nuclear fuel packages under hypothetical accident conditions	Not applicable to packages for fresh TRISO fuel transportation	Not applicable	Not applicable	Not applicable

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.4 Mechanical Properties	Tensile properties	Acceptability of material tensile properties	Mechanical properties for commonly used packaging materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that commonly used packaging materials may be also adequate for package design and fabrication for transport of TRISO fuel. If alternative or new materials were required in the design and fabrication of transportation packages, the SRP calls for examination of the adequacy of information in the application related to mechanical properties of those alternative materials.
	Fracture resistance	Acceptability of material fracture toughness	Mechanical properties for commonly used packaging materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Tensile properties and creep of aluminum alloys at elevated temperatures	Acceptability of the tensile properties and creep of aluminum alloys	Mechanical properties for commonly used aluminum alloys are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Impact limiters	Acceptability of the mechanical properties of the impact limiter materials	Mechanical properties for commonly used impact limiter materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.5 Thermal Properties of Materials		Thermal properties of package materials; Effect of degradation and anisotropic dependencies of thermal properties	Thermal properties for commonly used packaging materials are available; Detailed evaluation of package components and fuels is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.6 Radiation Shielding	Neutron-shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Gamma-shielding materials	Compositions and geometries of	Detailed evaluation of packaging materials is	None expected	None identified; General acceptance

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	expected to be provided by the applicant		criteria not impacted by specifics of technologies
7.4.7 Criticality Control	Neutron-absorbing (poison) material specification	Chemical composition, physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; Qualification testing	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Computation of percent credit for boron-based neutron absorbers	Level of credit allowed for absorber materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Qualifying properties not associated with attenuation	Qualification of absorber material properties not associated with	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		neutron attenuation			
7.4.8 Corrosion Resistance	Environments	Range of environmental conditions encountered for package components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Carbon and low-alloy steels	Environment dependencies of corrosion rate; Coatings for corrosion prevention	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Austenitic stainless steel	Localized corrosion and chloride-induced stress corrosion cracking in chloride-containing environments; Intergranular corrosion and stress corrosion cracking in sensitized stainless steel	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.9 Protective Coatings	Review guidance	Coating specifications	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Scope of coating application	Purpose of the coating, lists the components to be coated, and the expected environmental conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating selection	Coating manufacturer, type of primers and topcoat, coating thickness, and ability of the coating to withstand the in-service conditions	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating qualification testing	Qualification testing for coating performance in accordance with several standard ASTM (and possibly other) tests	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for evaluating any qualification testing for the demonstration of coating performance.
7.4.10 Content Reactions	Flammable and explosive reactions	Effects of flammable and explosive reactions among	Information available on NRC-certified Versa-Pac package (Century Industries,	None expected	None identified; General acceptance criteria not impacted

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		the content materials	2009, 2010); Detailed evaluation of package components and fuels is expected to be provided by the applicant		by specifics of technologies
	Content chemical reactions, outgassing, and corrosion	Effects of chemical reactions, outgassing, and corrosion among the contents and between the contents and the package components	Detailed evaluation of package components and fuels is expected to be provided by the applicant	None expected	Additional guidance may need to be developed to address corrosion of non-fuel hardware associated with TRISO fuel. The SRP calls for examining that corrosion wastage will not lead to a loss of intended functions; however, for non-fuel hardware the current review method is limited to guidance for the examination of corrosion of hardware components associated with stainless steel or zirconium alloy-clad UO_2 fuels.

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel				
Areas of review (NUREG-2216, Chapter 7)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.11 Radiation Effects	Effects of radiation on the performance of the package materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. Commonly used packaging materials may be also adequate for package design and fabrication for transport of TRISO fuel. If alternative or new materials were required in the design and fabrication of transportation packages, the SRP calls for examination of the adequacy of information in the application related to radiation effects on those alternative materials.

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel				
Areas of review (NUREG-2216, Chapter 7)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.12 Package Contents	Chemical and physical form of the package contents; Effects of corrosion, chemical reactions, and radiation on the properties of the contents	Detailed evaluation of package components and fuels is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.13 Fresh (Unirradiated) Fuel Cladding	Not applicable to fresh TRISO fuel	Not applicable	Not applicable	Additional guidance may need to be developed to address the mechanical properties of coating layers of TRISO fuel. The review method is limited to guidance for the examination of mechanical properties of zirconium and aluminum alloy cladding, which are not applicable to TRISO fuel. It is necessary to examine whether similar or equivalent cladding functions are required in TRISO fuel.

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.14 Spent Nuclear Fuel	Spent fuel classification	Not applicable to fresh TRISO fuel	Not applicable	Not applicable	Not applicable
	Uncanned spent fuel	Not applicable to fresh TRISO fuel	Not applicable	Not applicable	Not applicable
	Canned spent fuel	Not applicable to fresh TRISO fuel	Not applicable	Not applicable	Not applicable
7.4.15 Bolting Material		Material properties of the bolting; Effects of corrosion, chemical reactions, and radiation on the bolting materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.16 Seals	Metallic seals	Material properties of metallic seals; Effects of corrosion, chemical reactions, and radiation on the seal materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Elastomeric seals	Material properties of elastomeric seals; Effects of corrosion, chemical reactions, and	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		radiation on the seal materials			

Century Industries. "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container." Revision 0. ML092321070. Bristol, Virginia: Century Industries. 2009.

Century Industries. "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container." Revision 3. ML101110135. Bristol, Virginia: Century Industries. 2010.

NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

APPENDIX II

STORAGE OF SPENT FUEL

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent metal fuel

Areas of review (NUREG-2215, Chapter 4)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
4.5.1 Description of the Structures, Systems, and Components					
4.5.1.1 Structures, Systems, and Components Important to Safety	Canister or storage cask and metallic internals	Canister or storage cask design description	Information available on many NRC-certified storage container designs and previous experience of storing spent metal fuel in containers under wet and dry storage conditions at Idaho National Laboratory (INL, 2007); Detailed description of structural design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel basket	Fuel basket design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel and cladding	Fuel rod and cladding design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Transfer cask	Transfer cask design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Storage overpack (horizontal, vertical, or underground)	Storage overpack design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Independent spent fuel storage installations concrete pad (as applicable)	ISFSI concrete pad design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.1.2 Other Structures, Systems, and Components Subject to NRC Approval		Design descriptions of other SSCs subject to NRC approval	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.2 Design Criteria					
4.5.2.1 Structures, Systems, and Components Important to Safety	Canister and storage cask confinement shell	Canister or storage cask design criteria	Information available on many NRC-certified storage container designs; Detailed description of structural design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel basket	Fuel basket design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Fuel and cladding	Fuel rod and cladding design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Transfer cask	Transfer cask design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Storage overpack (horizontal, vertical, underground)	Storage overpack design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Independent spent fuel storage installations concrete pad	ISFSI concrete pad design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.2.2 Other Structures, Systems, and Components Subject to NRC Approval		Design descriptions of other SSCs subject to NRC approval	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.3 Loads	Normal conditions	Loads during normal conditions	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Off-normal conditions	Loads during off-normal conditions	Structural performance of the storage container	None expected	None identified; General acceptance

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 4)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
			under load conditions is expected to be provided by the applicant		criteria not impacted by specifics of technologies
	Accident conditions				
	Cask drop and tipover	Load conditions associated with cask drop and tipover	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Earthquake	Load conditions associated with earthquake	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tornado winds	Load conditions associated with tornado winds	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tornado missiles	Load conditions associated with tornado missiles	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Flood	Load conditions associated with flood	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fire	Load conditions associated with fire	Structural performance of the storage container under load conditions is	None expected	None identified; General acceptance criteria not impacted

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
			expected to be provided by the applicant		by specifics of technologies
	Explosive overpressure	Load conditions associated with explosive overpressure	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.4 Analytical Approach	Hand calculations	Structural analysis of various loading combinations using hand calculations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Finite element analyses	Structural analysis of various loading combinations using finite element analyses	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.5 Normal and Off-Normal Conditions					
4.5.5.1 Structures, Systems, and Components Important to Safety	Canister and associated welds and bolts	Structural analysis of canister and associated welds and bolts for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel basket	Structural analysis of fuel basket and associated welds for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Spent fuel assemblies and cladding	Structural analysis of spent fuel assemblies and	Structural analysis of various loading combinations is	None expected	None identified; General acceptance criteria not impacted

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		cladding for various loading combinations	expected to be provided by the applicant		by specifics of technologies
	Transfer cask	Structural analysis of transfer cask components for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Storage overpack	Structural analysis of steel and reinforced concrete structures for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.5.2 Other Structures, Systems, and Components Subject to NRC Approval		Structural analysis of other SSCs subject to NRC approval for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.6 Accident Conditions					
4.5.6.1 Structures, Systems, and Components Important to Safety	Canister and associated welds and bolts	Structural analysis of canister and associated welds and bolts for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel basket	Structural analysis of fuel basket and associated welds	Structural analysis of various loading combinations is	None expected	None identified; General acceptance criteria not impacted

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		for various loading combinations	expected to be provided by the applicant		by specifics of technologies
	Spent fuel assemblies and cladding	Structural analysis of spent fuel assemblies and cladding for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Transfer cask	Structural analysis of transfer cask components for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Storage overpack	Structural analysis of steel and reinforced concrete structures for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.6.2 Other Structures, Systems, and Components		Structural analysis of other SSCs subject to NRC approval for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
INL. “Idaho National Laboratory Preferred Disposition Plan for Sodium-Bonded Spent Nuclear Fuel.” Idaho Falls, Idaho: Idaho National Laboratory. 2007.					

Table II-2. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.5.1 Decay Heat Removal Systems	General considerations (SL)	Thermal design features and operating characteristics of all components under normal, loading, off-normal, and accident conditions	Information available on many NRC-certified storage container designs; Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Dry storage systems (SL)	Limiting conditions for operation and surveillance requirements	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Dry transfer systems (SL)	Limiting conditions for operation and surveillance requirements	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.5.2 Material and Design Limits	General considerations	Temperature limits for the fuel cladding	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Considerations for specific licenses (SL)	Temperature limits for the material of construction and the stored radioactive material	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-2. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.5.3 Thermal Loads and Environmental Conditions	General considerations	Thermal loading; Thermal impact of environmental conditions	Information available on many NRC-certified storage container designs; however, the decay heats of spent metal fuel and converted waste forms are uncertain	Specific decay heats of spent metal fuel and converted waste forms are to be evaluated	None identified. The review method calls for verification of the decay heat used in the thermal evaluation that is consistent with the specified fuel types, burnups, enrichments, and cooling times. The existing review method is sufficient to evaluate the decay heats of spent metal fuel and converted waste forms.
	Considerations for specific licenses (SL)	Thermal loading; Thermal impact of environmental conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-2. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.5.4 Analytical Methods, Models, and Calculations	Configuration	Description of the models in the thermal evaluation for normal, off-normal, and accident conditions	Specific models used to evaluate the storage container are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Material specifications and thermal properties for all components used in the models	Thermal properties for commonly used storage container materials are available. Some thermal properties of fuel pin components, structural components, and metal fuel are available (Leibowitz et al., 1976; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Data characterizing phases, phase diagrams, heat capacity, and thermal properties of metal fuel and converted waste forms are limited	None identified. The review method calls for verification of the thermal properties used in the thermal evaluation and potential degradation of materials over their service life. The existing review method is sufficient to deal with the assessment of required metal fuel properties important to the thermal analysis.
	Boundary conditions	Boundary conditions for normal, loading, off-normal, and accident conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-2. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent metal fuel

Areas of review (NUREG-2215, Chapter 5)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Computer codes	Computer codes used for thermal analysis	Codes used in the thermal analysis are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Temperature calculations	Maximum and minimum temperatures of all components under normal, loading, off-normal, and accident conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Pressure analysis	Method and assumptions used in the pressure analysis	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analysis	Rigor of the confirmatory analysis	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.5.5 Surveillance Requirements		Surveillance requirements related to thermal evaluations	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Janney, D.E. and S.L. Hayes. "Experimentally Known Properties of U-10Zr alloys: A critical Review." Nuclear Technology. Vol. 203. pp.109–128. 2018.
Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 1: Phases and Phase Diagrams." Nuclear Technology. Vol. 205. pp.1,387–1,415. 2019.
Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 2: Thermal and Mechanical Properties." Nuclear Technology. Vol. 206. pp.1–22. 2020.

Table II-2. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent metal fuel				
Areas of review (NUREG-2215, Chapter 5)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
Janney, D.E. "Metallic Fuels Handbook, Part 1: Alloys Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr, Including Those with Minor Actinides (Np, Am, Cm), Rare-earth Elements (La, Ce, Pr, Nd, Gd), and Y." INL/EXT-15-36520 Revision 3 Part 1. Idaho Falls, Idaho: Idaho National Laboratory. 2018a. _____. "Metallic Fuels Handbook, Part 2: Elements and Alloys not Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr." INL/EXT-15-36520 Revision 3 Part 2. Idaho Falls, Idaho: Idaho National Laboratory. 2018b. Leibowitz, L., E.C. Chang, M.G. Chasanov, R.L. Gibby, C. Kim, A.C. Millunzi, D. Stahl. "Properties for Liquid Metal Fast Breeder Reactor Safety Analysis." Argonne National Laboratory. ANL-CEN-RSD-76-1. 1976.				

Table II-3. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems and facilities for spent metal fuel

Areas of review (NUREG-2215, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
6.5.1 Shielding Design Description	Design criteria	Shielding design criteria	Information available on many NRC-certified storage container designs and previous experience of storing spent metal fuel in containers under wet and dry storage conditions at Idaho National Laboratory (INL, 2007); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Design features	Shielding design features	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.5.2 Radiation Source Definition	Initial enrichment	Minimum initial enrichment used in the criticality analysis	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Computer codes for radiation source definition	Computer codes used to determine the radiological and thermal source terms for the shielding analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-3. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Gamma sources	Gamma source strengths and spectra for the contents and activated hardware	Metal fuel fabricated with reprocessed uranium could possibly contain radioactive sources; Although ASTM C996 (ASTM, 2021) is frequently applied, information in the literature is limited	Source-term specification including gamma sources and their energies arising from fission products from reprocessed uranium or other impurities is to be evaluated	None identified. The review method calls for verifying the gamma source terms as a function of energy to ensure that the total source is correctly considered in the shielding evaluation. The existing review method is sufficient to evaluate residual gamma sources in metal fuel fabricated with reprocessed uranium.
	Neutron sources	Neutron source strengths and spectra for the contents and activated hardware	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Other parameters affecting the source term	Information concerning reactor operations that affect the source terms for the shielding analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-3. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems and facilities for spent metal fuel

Areas of review (NUREG-2215, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
6.5.3 Shielding Model Specification	Configuration of shielding and source	Geometric arrangements and physical dimensions of the storage container components and shielding features; Source term locations and physical distribution and material properties of the sources	Information available on many NRC-certified storage container designs; Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Material compositions and densities used in the shielding models	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.5.4 Shielding Analyses	Computer codes	Applicability and appropriateness of the codes used in the shielding evaluation for normal, off-normal, and accident conditions	Codes used for shielding evaluation are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Flux-to-dose-rate conversion	Accuracy and acceptance of the conversion factors	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-3. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Dose rates	Dose rates and their variation with locations	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.5.5 Consideration of Reactor-Related GTCC Waste Storage (SL)		Not applicable to spent metal fuel	Not applicable	Not applicable	Not applicable

ASTM. C996. "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5 percent ^{235}U" <https://compass.astm.org/EDIT/html_annot.cgi?C996+20> (Accessed May 27, 2021). 2021.
INL. "Idaho National Laboratory Preferred Disposition Plan for Sodium-Bonded Spent Nuclear Fuel." Idaho Falls, Idaho: Idaho National Laboratory. 2007.

Table II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.5.1 Criticality Design Criteria and Features		Criticality design criteria; Design features significant to the criticality design	Information available on many NRC-certified storage container designs; Detailed criticality design criteria and features are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.5.2 Fuel Specification	Fuel type	Specifications for the ranges or types of spent fuel, including type of fuel assemblies, maximum fuel enrichment, and fuel density	Information available on spent metal fuels stored in stainless steel containers under wet and dry storage conditions at Idaho National Laboratory (INL, 2007; NWTRB, 2017; Pahl, 2000; Pahl et al., 1996); Detailed fuel specifications are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Nonfuel hardware	Identification of nonfuel components; Effects of both inclusion and neglect of nonfuel hardware on system reactivity	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Fuel condition	Classification of damaged, undamaged, and intact fuel	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.5.3 Model Specification	Configuration	Description of the criticality models used to evaluate normal, off-normal, and accident conditions	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Compositions and densities for all materials used in the criticality models	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.5.4 Criticality Analysis	Computer codes and cross-section data	Computer codes and cross-section data used for criticality calculations	Codes and cross-section data used in the criticality analysis are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Neutron multiplication factor	Results of the k_{eff} calculations; Independent analysis of the criticality calculations	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Benchmark comparisons	Benchmarking computer codes for criticality calculations against critical experiments; Description of the benchmark comparisons	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001); however, criticality benchmark data and validation of existing criticality codes and methods are limited for enrichments between 5 and 20 weight percent (Jarrell, 2018).	Criticality benchmark data and applicability of existing criticality codes and methods for spent metal fuel with enrichments between 5 and 20 weight percent ^{235}U are to be evaluated	None identified. The review method calls for a thorough comparison to justify the validity of computer codes for criticality calculations that have been benchmarked against critical experiments applicable to the actual storage container design and contents. The existing review method is sufficient to deal with the availability of criticality benchmark data and applicability of existing criticality codes and methods for spent metal fuel.

Table II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent metal fuel

Areas of review (NUREG-2215, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.5.5 Burnup Credit	Limits for the licensing basis	Analytic methods, assumptions, and assay data used in the burnup credit analyses for the licensing basis	Metal fuel has an enrichment of 26–93 weight percent ^{235}U and a fuel burnup of 38–143 GWd/MTU (FRWG, 2018)	Burnup credit analyses for spent metal fuel storage container designs with increasing enrichment and fuel burnup limits are to be evaluated, given the potential lack of code validation data	Additional guidance may need to be developed to address storage of high burnup and enriched metal fuel. The review method specifies the current licensed fuel burnup and enrichment limits for storing light water reactor fuel (i.e., 60 GWd/MTU burnup and 5.0 weight percent ^{235}U enrichment).
	Licensing-basis model assumptions	Models and analysis assumptions for the k_{eff} calculations representative of the physics in the storage container	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Code validation—isotopic depletion	Validation of the depletion codes; Bias and bias uncertainty of the codes	Metal fuel has an enrichment of 26–93 weight percent ^{235}U and a fuel burnup of 38–143 GWd/MTU (FRWG, 2018)	Depletion analyses for spent metal fuel storage container designs with increasing enrichment and fuel burnup limits are to be evaluated, given the potential lack of code validation data	Additional guidance may need to be developed to address storage of high burnup and enriched metal fuel. The review method is limited to burnup credit available from actinide compositions associated with UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated in a pressurized-water reactor to an assembly-average burnup value not exceeding 60 GWd/MTU.

Table II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Code validation—k_{eff} determination	Bias and bias uncertainty associated with actinide-only, and fission product and minor actinide burnup credit	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001); however, criticality benchmark data and validation of existing criticality codes and methods are limited for enrichments between 5 and 20 weight percent (Jarrell, 2018).	Criticality benchmarking for spent metal fuel with enrichments between 5 and 20 weight percent ^{235}U is to be evaluated	Additional guidance may need to be developed to address transportation of high burnup and enriched metal fuel. The review method is limited to burnup credit available from actinide and fission product compositions associated with UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated in a pressurized-water reactor to an assembly-average burnup value not exceeding 60 GWd/MTU.
	Loading curve and burnup verification	Burnup credit loading curves; Performance of burnup verification	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.5.6 Reactor-Related Greater-Than-Class-C Waste and HLW (SL)		Not applicable because of irrelevant wastes	Not applicable	Not applicable	Not applicable
7.5.7 Supplemental Information		Not applicable	Not applicable	Not applicable	Not applicable

Table II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent metal fuel				
Areas of review (NUREG-2215, Chapter 7)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
ANS. American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998 (R2007). "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." La Grange Park, Illinois: American Nuclear Society. 2007. FRWG. "Nuclear Metal Fuel: Characteristics, Design, Manufacturing, Testing, and Operating History." White Paper 18-01. ML18165A249. Fast Reactor Working Group. 2018. INL. "Idaho National Laboratory Preferred Disposition Plan for Sodium-Bonded Spent Nuclear Fuel." Idaho Falls, Idaho: Idaho National Laboratory. 2007. Jarrell, J. "A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium." INL/EXT-18-51518. Idaho Falls, Idaho: Idaho National Laboratory. 2018. NRC. NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology." Oak Ridge, Tennessee: Science Applications International Corporation. U.S. Nuclear Regulatory Commission. 2001. _____. NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages." ORNL/TM-13211. Oak Ridge, Tennessee: Oak Ridge National Laboratory. U.S. Nuclear Regulatory Commission. 1997. NWTRB. "Management and Disposal of U.S. Department of Energy Spent Nuclear Fuel." Washington, DC: Nuclear Waste Technical Review Board. 2017. Pahl, R.G. "Characterization of Degraded EBR-II Fuel from the ICPP-603 Basin: National Spent Nuclear Fuel Program FY1999 Final Report." Idaho Falls, Idaho: Argonne National Laboratory. 2000. Pahl, R.G., E.M. Franklin, and M.A. Ebner. "Technical Assessment of Continued Wet Storage of EBR-II Fuel." Reno, Nevada: DOE Spent Nuclear Fuel and Fissile Material Management Conference. 1996.				

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel

Areas of review (NUREG-2215, Chapter 8)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
8.5.1 Drawings		Content of drawings	Information available on many NRC-certified storage container designs; Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for examining the content of engineering drawings as well as the description of materials in storage container designs.
8.5.2 Codes and Standards	Usage and endorsement	Codes and standards used for the storage container design and construction	Codes and standards are available, but not specifically for new materials to be used for storage container design and fabrication for storage of metal fuel	Applicability of codes and standards for storage container design and fabrication with new materials needs to be evaluated	None identified. The review method calls for verification of the codes and standards for storage container components important to safety. Codes and standards to be used are expected to be defined or developed, or the technical basis be provided on the adequacy of alternative codes and standards.
	Code case use and acceptability	Acceptability of ASME code cases	Specific code case referenced is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
8.5.3 Welding	Confinement weld design	Welding criteria and weld procedure qualification requirements	Detailed weld design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that standard welding processes are adequate for storage container design and fabrication for storage of metal fuel. If new technologies were used in the design and fabrication of welds, the SRP calls for examination of compliance with any established codes and standards proposed in the application for design and construction.
	Confinement weld inspection	Weld inspection methods and requirements	Detailed weld inspection information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Confinement weld testing	Weld testing methods and requirements	Detailed weld testing information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.4 Mechanical Properties of Metals	Tensile properties	Acceptability of material tensile properties	Mechanical properties for commonly used storage container materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that commonly used materials are adequate for storage container design and fabrication for storage of metal fuel. If alternative or new materials were required in the design and fabrication of storage containers, the SRP calls for examination of the adequacy of information in the application related to the mechanical properties of those alternative materials.

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Fracture resistance	Acceptability of material fracture toughness	Mechanical properties for commonly used storage container materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Performance of aluminum components	Acceptability of the tensile properties, fracture toughness, and creep of aluminum materials	Mechanical properties for commonly used aluminum materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.5 Thermal Properties		Thermal properties of storage container materials; Effect of degradation and anisotropic dependencies of thermal properties	Thermal properties for commonly used storage container materials are available; Some thermal properties of fuel pin components, structural components, and metal fuel are available (Leibowitz et al., 1976; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Data characterizing phases, phase diagrams, heat capacity, and thermal properties of metal fuel and converted waste forms are limited	None identified. The review method calls for verifying the thermal properties and the change in these properties from material degradation. The existing review method is sufficient to deal with the availability of information related to metal fuel properties important to the thermal analysis.

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
8.5.6 Radiation Shielding Materials	Neutron shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Assessing previously unreviewed (new) neutron shielding materials	Temperature and radiation-induced degradation and its effects on neutron shield performance	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Gamma shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
8.5.7 Criticality Control Materials	Neutron absorbing (poison) material specification	Chemical composition, physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; Qualification testing	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Computation of percent credit for boron-based neutron absorbers	Level of credit allowed for absorber materials	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Qualifying the neutron absorber material fabrication process	Qualification of absorber material properties not associated with neutron attenuation	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel

Areas of review (NUREG-2215, Chapter 8)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
8.5.8 Concrete and Reinforcing Steel	Embedment materials	Material used for embedments, inserts, conduits, pipes, or other items embedded in the concrete	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Concrete design and temperature limits	Design and material specifications for the concrete; Temperature requirements; Changes in concrete properties with time	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Omission of reinforcement	Omission of reinforcing steel in the concrete	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Radiation damage	Radiation effects on concrete properties	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.9 Bolt Applications		Material properties of the bolting; Effect of corrosion on the bolting materials; Closure bolt stresses	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
8.5.10 Seals	Metallic seals	Material properties of metallic seals	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Elastomeric seals	Material properties of elastomeric seals; Effects of thermal, radiation, and chemical reactions on the seal materials	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.11 Corrosion Resistance	Environments	Range of environmental conditions encountered for storage container components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Carbon and low-alloy steels	Environment dependencies of corrosion rate; Coatings for corrosion prevention	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Austenitic stainless steels	Localized corrosion and stress corrosion cracking in chloride-containing environments; Chloride-induced stress corrosion cracking in sensitized stainless steels	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Duplex stainless steels	Microstructural alteration in welded duplex stainless steels; Fabrication and weld testing and acceptance criteria	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.12 Protective Coatings	Review guidance	Coating specifications	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Scope of coating application	Purpose of the coating, lists the components to be coated, and the expected environmental conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Coating selection	Coating manufacturer, type of primers and topcoat, coating thickness, and ability of the coating to withstand the in-service conditions	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating qualification testing	Qualification testing for coating performance in accordance with several standard ASTM (and possibly other) tests	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for evaluating any qualification testing for the demonstration of coating performance.

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
8.5.13 Content Reactions	Flammable and explosive reactions	Effects of flammable and explosive reactions among the content materials	Sodium reacts violently with water, which produces sodium hydroxide and hydrogen, and the hydrogen burns when in contact with air	Safety protocols in storing sodium-containing metal fuel are to be established	None identified. The review method calls for measures to remove moisture for detecting the presence of hydrogen and preventing the ignition of combustible gases. The existing review method is sufficient to evaluate the effects of reactions of sodium and fuel with water in the context of storage of spent metal fuel.
	Corrosion	Effects of corrosive reactions among the contents and between the contents and the storage container components	Some material properties and performance data are available on metal fuel (Carmack et al., 2009; Garner, 1993; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Effects of air and water on chemical interactions, including reactions with sodium in the fuel pin and galvanic coupling of storage container internal materials, are to be evaluated	Additional guidance may need to be developed to address the corrosion of non-fuel hardware associated with metal fuel. The SRP calls for examining whether corrosion wastage could lead to a loss of intended functions; however, for non-fuel hardware, the current review method is limited to guidance for the examination of

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
					corrosion of hardware components associated with stainless steel or zirconium alloy-clad UO_2 fuels.
8.5.14 Management of Aging Degradation	Initial storage term	Materials performance of storage container components for the duration of the storage term	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Amendment applications submitted during a renewal review or after a renewal is issued	Aging management for the amendment applications	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.15 Spent Fuel	Spent fuel classification	Classification of damaged, undamaged, and intact fuel	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Uncannistered spent fuel	Cladding alloys and maximum fuel burnup; Cladding mechanical properties; Effective cladding thickness; Maximum cladding	Stainless steel cladding performance in storage environments may be challenged by sensitization, intergranular attack, stress corrosion cracking, thermal	Performance of stainless steel cladding and sodium-bonded spent metal fuel under storage environments needs to be	Additional guidance may need to be developed to address mechanical properties of stainless steel and advanced cladding materials for metal fuel. The current

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		temperature; Thermal cycling during loading operations; Composition of the cover gas; High burnup fuel monitoring and assessment; Release fractions	aging, and radiation embrittlement (Alexander and Nanstand, 1995; Chandra et al., 2012; Guenther et al., 1996). Sodium-bonded spent metal fuel may experience degradation during storage, particularly oxidation, hydriding, fragmentation, and restructuring-swelling (Guenther et al., 1996).	evaluated; Advanced cladding material properties that can be used to achieve high burnup are to be evaluated, especially material performance data under the influence of irradiation	review method is limited to guidance for the mechanical properties of zirconium alloy cladding.
	Cannistered spent fuel	Performance of the fuel can for damaged fuel	Detailed evaluation of the fuel can performance is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Alexander, D.J. and R.K. Nanstand. "The Effects of Aging for 50,000 Hours at 343°C on the Mechanical Properties of Type 308 Stainless Steel Weldments." Proceedings of the Seventh International Symposium on Environmental Degradation of Materials in Nuclear Power Systems–Water Reactors. Breckenridge, Colorado. NACE. Houston, Texas. pp. 747–758. 1995.

Carmack, W., D. Porter, Y.H.S. Chang, M. Meyer, D. Burkes, C. Lee, T. Mizuno, F. Delage, and J. Somers. "Metallic Fuels for Advanced Reactors." *Journal of Nuclear Materials.* Vol. 392. pp. 139–150. 2009.

Chandra, K., K. Vivekanand, V.S. Raja, R. Tewari, and G.K. Dey. "Low Temperature Thermal Ageing Embrittlement of Austenitic Stainless Steel Welds and its Electrochemical Assessment." Corrosion Science. Vol. 54. pp. 278–290. 2012.

Garner, F.A. "Irradiation Performance of Cladding and Structural Steels in Liquid Metal Reactors." Nuclear Materials: Part 1. Materials Science and Technology: A Comprehensive Treatment. Frost, B.R.T., Editor. VCH Publishers. pp. 419–543. 1993.

Guenther, R.J., A.B. Johnson, A.L. Lund, E.R. Gilbert, S.P. Pednekar, F.M. Berting, L.L. Burger, S.A. Bryan, and T.M. Orlando. "Initial Evaluation of Dry Storage Issues for Spent Nuclear Fuels in Wet Storage at the Idaho Chemical Processing Plant." INEL-96/0140. Idaho National Engineering Laboratory. 1996.

IAEA. "Structural Materials for Liquid Metal Cooled Fast Reactor Fuel Assemblies: Operational Behaviour." Nuclear Energy Series No. NF-T-4.3. Vienna, Austria: International Atomic Energy Agency. 2012.

Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent metal fuel

Areas of review (NUREG-2215, Chapter 8)	Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
Janney, D.E. “Metallic Fuels Handbook, Part 1: Alloys Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr, Including Those with Minor Actinides (Np, Am, Cm), Rare-earth Elements (La, Ce, Pr, Nd, Gd), and Y.” INL/EXT-15-36520 Revision 3 Part 1. Idaho Falls, Idaho: Idaho National Laboratory. 2018a. _____. “Metallic Fuels Handbook, Part 2: Elements and Alloys not Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr.” INL/EXT-15-36520 Revision 3 Part 2. Idaho Falls, Idaho: Idaho National Laboratory. 2018b. Janney, D.E. and S.L. Hayes. “Experimentally Known Properties of U-10Zr alloys: A critical Review.” *Nuclear Technology*. Vol. 203. pp.109–128. 2018. Janney, D.E., S.L. Hayes, and C.A. Adkins. “A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 1: Phases and Phase Diagrams.” *Nuclear Technology*. Vol. 205. pp.1,387–1,415. 2019. Janney, D.E., S.L. Hayes, and C.A. Adkins. “A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 2: Thermal and Mechanical Properties.” *Nuclear Technology*. Vol. 206. pp.1–22. 2020. Leibowitz, L., E.C. Chang, M.G. Chasanov, R.L. Gibby, C. Kim, A.C. Millunzi, and D. Stahl. “Properties for Liquid Metal Fast Breeder Reactor Safety Analysis.” ANL-CEN-RSD-76-1. Lemont, Illinois: Argonne National Laboratory. 1976.				

Table II-6. Information to be reviewed and potential gaps for evaluating confinement performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 9)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
9.5.1 Confinement Design Characteristics	Design criteria	Confinement design criteria	Information available on many NRC-certified storage container designs and previous experience of storing spent metal fuel in containers under wet and dry storage conditions at Idaho National Laboratory (INL, 2007); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Design features	Confinement design features	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
9.5.2 Confinement Monitoring Capability		Leakage test, monitoring systems, and surveillance requirements of the storage container	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
9.5.3 Nuclides with Potential for Release		Availability and release fractions of radioactive nuclides	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-6. Information to be reviewed and potential gaps for evaluating confinement performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 9)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
9.5.4 Confinement Analyses	Normal conditions	Confinement analysis and the resulting doses for the normal conditions	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Off-normal conditions (anticipated occurrences)	Confinement analysis and the resulting doses for the off-normal conditions	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Design-basis accident conditions (including natural phenomenon events)	Confinement analysis and the resulting doses for the accident conditions	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Identification of release events (SL)	Spectrum of release events for normal operations, off-normal operations, and design-basis accidents	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-6. Information to be reviewed and potential gaps for evaluating confinement performance of dry storage systems and facilities for spent metal fuel					
Areas of review (NUREG-2215, Chapter 9)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Evaluation of release estimates for spent nuclear fuel and high-level radioactive waste (SL)	Dose calculations and release estimates for normal operations, off-normal operations, and design-basis accidents	Information available on many NRC-certified storage container designs and previous experience of breached spent metal fuel stored in stainless steel containers under wet and dry storage conditions at Idaho National Laboratory (DOE, 2012; Pahl, 2000; Pahl et al., 1996); Detailed confinement evaluation is expected to be provided by the applicant	Confinement performance of sodium-bonded metal fuel and reactive nature of metallic sodium in fuel are to be evaluated	None identified. The review method calls for confinement analyses to demonstrate compliance with the dose limits specified in 10 CFR 72.104(a) and 72.106(b). The existing review method is sufficient to evaluate the confinement performance of spent metal fuel.
	Evaluation of release estimates for reactor-related greater than Class C waste (SL)	Not applicable to spent metal fuel	Not applicable	Not applicable	Not applicable
9.5.5 Supplemental Information		Not applicable	Not applicable	Not applicable	Not applicable

DOE. “Strategy for Disposition of Non-EMT Candidate Sodium Bonded Driver Fuels.” TEM-10300-1. Rev. 3. 2012.

INL. “Idaho National Laboratory Preferred Disposition Plan for Sodium-Bonded Spent Nuclear Fuel.” Idaho Falls, Idaho: Idaho National Laboratory. 2007.

Pahl, R.G. “Characterization of Degraded EBR-II Fuel from the ICPP-603 Basin: National Spent Nuclear Fuel Program FY1999 Final Report.” Idaho Falls, Idaho: Argonne National Laboratory. 2000.

Pahl, R.G., E.M. Franklin, and M.A. Ebner. “Technical Assessment of Continued Wet Storage of EBR-II Fuel.” Reno, Nevada: DOE Spent Nuclear Fuel and Fissile Material Management Conference. 1996.

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
4.5.1 Description of the Structures, Systems, and Components					
4.5.1.1 Structures, Systems, and Components Important to Safety	Canister or storage cask and metallic internals	Canister or storage cask design description	Information available on many NRC-certified storage container designs, particularly the design of the Fort St. Vrain ISFSI for storage of spent TRISO fuel (DOE, 2010; NRC, 2011); Detailed description of structural design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel basket	Fuel basket design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel and cladding	Fuel rod and cladding design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Transfer cask	Transfer cask design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Storage overpack (horizontal, vertical, or underground)	Storage overpack design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Independent spent fuel storage installations concrete pad (as applicable)	ISFSI concrete pad design description	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.1.2 Other Structures, Systems, and Components Subject to NRC Approval		Design descriptions of other SSCs subject to NRC approval	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.2 Design Criteria					
4.5.2.1 Structures, Systems, and Components Important to Safety	Canister and storage cask confinement shell	Canister or storage cask design criteria	Information available on many NRC-certified storage container designs; Detailed description of structural design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel basket	Fuel basket design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Fuel and cladding	Fuel rod and cladding design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Transfer cask	Transfer cask design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Storage overpack (horizontal, vertical, underground)	Storage overpack design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Independent spent fuel storage installations concrete pad	ISFSI concrete pad design criteria	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.2.2 Other Structures, Systems, and Components Subject to NRC Approval		Design descriptions of other SSCs subject to NRC approval	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.3 Loads	Normal conditions	Loads during normal conditions	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Off-normal conditions	Loads during off-normal conditions	Structural performance of the storage container	None expected	None identified; General acceptance

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
			under load conditions is expected to be provided by the applicant		criteria not impacted by specifics of technologies
	Accident conditions				
	Cask drop and tipover	Load conditions associated with cask drop and tipover	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Earthquake	Load conditions associated with earthquake	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tornado winds	Load conditions associated with tornado winds	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tornado missiles	Load conditions associated with tornado missiles	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Flood	Load conditions associated with flood	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fire	Load conditions associated with fire	Structural performance of the storage container under load conditions is	None expected	None identified; General acceptance criteria not impacted

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
			expected to be provided by the applicant		by specifics of technologies
	Explosive overpressure	Load conditions associated with explosive overpressure	Structural performance of the storage container under load conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.4 Analytical Approach	Hand calculations	Structural analysis of various loading combinations using hand calculations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Finite element analyses	Structural analysis of various loading combinations using finite element analyses	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.5 Normal and Off-Normal Conditions					
4.5.5.1 Structures, Systems, and Components Important to Safety	Canister and associated welds and bolts	Structural analysis of canister and associated welds and bolts for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel basket	Structural analysis of fuel basket and associated welds for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Spent fuel assemblies and cladding	Structural analysis of spent fuel assemblies and	Structural analysis of various loading combinations is	None expected	None identified; General acceptance criteria not impacted

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		cladding for various loading combinations	expected to be provided by the applicant		by specifics of technologies
	Transfer cask	Structural analysis of transfer cask components for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Storage overpack	Structural analysis of steel and reinforced concrete structures for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.5.2 Other Structures, Systems, and Components Subject to NRC Approval		Structural analysis of other SSCs subject to NRC approval for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.6 Accident Conditions					
4.5.6.1 Structures, Systems, and Components Important to Safety	Canister and associated welds and bolts	Structural analysis of canister and associated welds and bolts for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel basket	Structural analysis of fuel basket and associated welds	Structural analysis of various loading combinations is	None expected	None identified; General acceptance criteria not impacted

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		for various loading combinations	expected to be provided by the applicant		by specifics of technologies
	Spent fuel assemblies and cladding	Structural analysis of spent fuel assemblies and cladding for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Transfer cask	Structural analysis of transfer cask components for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Storage overpack	Structural analysis of steel and reinforced concrete structures for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.5.6.2 Other Structures, Systems, and Components		Structural analysis of other SSCs subject to NRC approval for various loading combinations	Structural analysis of various loading combinations is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
DOE. “Safety Analysis Report for Fort St. Vrain Independent Spent Fuel Storage Installation.” Revision 8. Chapter 3. ML103640368. Idaho Falls, Idaho: U.S. Department of Energy Idaho Operations Office. 2010. NRC. “Safety Evaluation Report for License Renewal: Fort St. Vrain Independent Spent Fuel Storage Installation.” ML112000261. Washington, DC: U.S. Nuclear Regulatory Commission. 2011.					

Table II-8. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.5.1 Decay Heat Removal Systems	General considerations (SL)	Thermal design features and operating characteristics of all components under normal, loading, off-normal, and accident conditions	Information available on many NRC-certified storage container designs, particularly the design of the Fort St. Vrain ISFSI for storage of spent TRISO fuel (DOE, 2010; NRC, 2011); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Dry storage systems (SL)	Limiting conditions for operation and surveillance requirements	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Dry transfer systems (SL)	Limiting conditions for operation and surveillance requirements	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-8. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.5.2 Material and Design Limits	General considerations	Temperature limits for the fuel cladding	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Considerations for specific licenses (SL)	Temperature limits for the material of construction and the stored radioactive material	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.5.3 Thermal Loads and Environmental Conditions	General considerations	Thermal loading; Thermal impact of environmental conditions	Information available on the NRC-certified storage container design of the Fort St. Vrain ISFSI for storage of spent TRISO fuel (DOE, 2010; NRC, 2011); however, the TRISO fuel discharged from fluoride salt-cooled high-temperature reactors (FHR) is expected to have much higher decay heat than that from high-temperature gas-cooled reactors (Forsberg and Peterson, 2015)	Higher decay heat with FHR fuel compared to that stored at the Fort St. Vrain ISFSI needs to be evaluated	None identified. The review method calls for verification of the decay heat used in the thermal evaluation that is consistent with the specified fuel types, burnups, enrichments, and cooling times. The existing review method is sufficient to evaluate the FHR fuel with higher decay heat.

Table II-8. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Considerations for specific licenses (SL)	Thermal loading; Thermal impact of environmental conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.5.4 Analytical Methods, Models, and Calculations	Configuration	Description of the models in the thermal evaluation for normal, off-normal, and accident conditions	Specific models used to evaluate the storage container are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Material specifications and thermal properties for all components used in the models	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Boundary conditions	Boundary conditions for normal, loading, off-normal, and accident conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Computer codes	Computer codes used for thermal analysis	Codes used in the thermal analysis are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-8. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 5)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Temperature calculations	Maximum and minimum temperatures of all components under normal, loading, off-normal, and accident conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Pressure analysis	Method and assumptions used in the pressure analysis	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analysis	Rigor of the confirmatory analysis	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.5.5 Surveillance Requirements		Surveillance requirements related to thermal evaluations	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

DOE. “Safety Analysis Report for Fort St. Vrain Independent Spent Fuel Storage Installation.” Revision 8. Chapter 3. ML103640368. Idaho Falls, Idaho: U.S. Department of Energy Idaho Operations Office. 2010.
Forsberg, C. and P.F. Peterson. “Spent Nuclear Fuel and Graphite Management for Salt-Cooled Reactors: Storage, Safeguards, and Repository Disposal.” Nuclear Technology. Vol. 191. pp. 113–121. 2015.
NRC. “Safety Evaluation Report for License Renewal: Fort St. Vrain Independent Spent Fuel Storage Installation.” ML112000261. Washington, DC: U.S. Nuclear Regulatory Commission. 2011.

Table II-9. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
6.5.1 Shielding Design Description	Design criteria	Shielding design criteria	Information available on many NRC-certified storage container designs, particularly the design of the Fort St. Vrain ISFSI for storage of spent TRISO fuel (DOE, 2010; NRC, 2011); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Design features	Shielding design features	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.5.2 Radiation Source Definition	Initial enrichment	Minimum initial enrichment used in the criticality analysis	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Computer codes for radiation source definition	Computer codes used to determine the radiological and thermal source terms for the shielding analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-9. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems and facilities for spent TRISO fuel

Areas of review (NUREG-2215, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Gamma sources	Gamma source strengths and spectra for the contents and activated hardware	TRISO fuel fabricated with reprocessed uranium could possibly contain radioactive sources; Although ASTM C996 (ASTM, 2021) is frequently applied, information in the literature is limited	Source-term specification including gamma sources and their energies arising from fission products from reprocessed uranium or other impurities is to be evaluated	None identified. The review method calls for verifying the gamma source terms as a function of energy to ensure that the total source is correctly considered in the shielding evaluation. The existing review method is sufficient to evaluate residual gamma sources in TRISO fuel fabricated with reprocessed uranium.
	Neutron sources	Neutron source strengths and spectra for the contents and activated hardware	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Other parameters affecting the source term	Information concerning reactor operations that affect the source terms for the shielding analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-9. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
6.5.3 Shielding Model Specification	Configuration of shielding and source	Geometric arrangements and physical dimensions of the storage container components and shielding features; Source term locations and physical distribution and material properties of the sources	Information available on many NRC-certified storage container designs; Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Material compositions and densities used in the shielding models	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.5.4 Shielding Analyses	Computer codes	Applicability and appropriateness of the codes used in the shielding evaluation for normal, off-normal, and accident conditions	Codes used for shielding evaluation are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Flux-to-dose-rate conversion	Accuracy and acceptance of the conversion factors	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-9. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Dose rates	Dose rates and their variation with locations	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.5.5 Consideration of Reactor-Related GTCC Waste Storage (SL)		Not applicable to spent TRISO fuel	Not applicable	Not applicable	Not applicable

ASTM. C996. "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5 percent ^{235}U" <https://compass.astm.org/EDIT/html_annot.cgi?C996+20> (Accessed May 27, 2021). 2021.
DOE. "Safety Analysis Report for Fort St. Vrain Independent Spent Fuel Storage Installation." Revision 8. Chapter 3. ML103640368. Idaho Falls, Idaho: U.S. Department of Energy Idaho Operations Office. 2010.
NRC. "Safety Evaluation Report for License Renewal: Fort St. Vrain Independent Spent Fuel Storage Installation." ML112000261. Washington, DC: U.S. Nuclear Regulatory Commission. 2011.

Table II-10. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent TRISO fuel

Areas of review (NUREG-2215, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.5.1 Criticality Design Criteria and Features		Criticality design criteria; Design features significant to the criticality design	Information available on many NRC-certified storage container designs, particularly the design of the Fort St. Vrain ISFSI for storage of spent TRISO fuel (DOE, 2010; NRC, 2011); Detailed criticality design criteria and features are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.5.2 Fuel Specification	Fuel type	Specifications for the ranges or types of spent fuel, including type of fuel assemblies, maximum fuel enrichment, and fuel density	Information available on spent TRISO fuels stored in containers at the Fort St. Vrain ISFSI and Arbeitsgemeinschaft Versuchsreaktor (IAEA, 2012, 1988; NWTRB, 2017); Detailed fuel specifications are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-10. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Nonfuel hardware	Identification of nonfuel components; Effects of both inclusion and neglect of nonfuel hardware on system reactivity	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fuel condition	Classification of damaged, undamaged, and intact fuel	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.5.3 Model Specification	Configuration	Description of the criticality models used to evaluate normal, off-normal, and accident conditions	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Compositions and densities for all materials used in the criticality models	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.5.4 Criticality Analysis	Computer codes and cross-section data	Computer codes and cross-section data used for criticality calculations	Codes and cross-section data used in the criticality analysis are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-10. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Neutron multiplication factor	Results of the k_{eff} calculations; Independent analysis of the criticality calculations	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Benchmark comparisons	Benchmarking computer codes for criticality calculations against critical experiments; Description of the benchmark comparisons	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001); however, criticality benchmark data and validation of existing criticality codes and methods are limited for enrichments between 5 and 20 weight percent (Jarrell, 2018).	Criticality benchmark data and applicability of existing criticality codes and methods for spent TRISO fuel with enrichments between 5 and 20 weight percent ^{235}U are to be evaluated	None identified. The review method calls for a thorough comparison to justify the validity of computer codes for criticality calculations that have been benchmarked against critical experiments applicable to the actual storage container design and contents. The existing review method is sufficient to deal with the availability of criticality benchmark data and applicability of existing criticality codes and methods for spent TRISO fuel.

Table II-10. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.5.5 Burnup Credit	Limits for the licensing basis	Analytic methods, assumptions, and assay date used in the burnup credit analyses for the licensing basis	TRISO fuel has an enrichment up to 20 weight percent ^{235}U and a fuel burnup of 150–200 GWd/MTU (NEA, 2014)	Burnup credit analyses for spent TRISO fuel storage container designs with increasing enrichment and fuel burnup limits are to be evaluated, given the potential lack of code validation data	Additional guidance may need to be developed to address storage of high burnup and enriched TRISO fuel. The review method specifies the current licensed fuel burnup and enrichment limits for storing light water reactor fuel (i.e., 60 GWd/MTU burnup and 5.0 weight percent ^{235}U enrichment).
	Licensing-basis model assumptions	Models and analysis assumptions for the k_{eff} calculations representative of the physics in the storage container	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-10. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Code validation—isotopic depletion	Validation of the depletion codes; Bias and bias uncertainty of the codes	TRISO fuel has an enrichment up to 20 weight percent ^{235}U and a fuel burnup of 150–200 GWd/MTU (NEA, 2014)	Depletion analyses for spent TRISO fuel storage container designs with increasing enrichment and fuel burnup limits are to be evaluated, given the potential lack of code validation data	Additional guidance may need to be developed to address storage of high burnup and enriched TRISO fuel. The review method is limited to burnup credit available from actinide compositions associated with UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated in a pressurized-water reactor to an assembly-average burnup value not exceeding 60 GWd/MTU.

Table II-10. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent TRISO fuel

Areas of review (NUREG-2215, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Code validation—k_{eff} determination	Bias and bias uncertainty associated with actinide-only, and fission product and minor actinide burnup credit	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001); however, criticality benchmark data and validation of existing criticality codes and methods are limited for enrichments between 5 and 20 weight percent (Jarrell, 2018).	Criticality benchmarking for spent TRISO fuel with enrichments between 5 and 20 weight percent ^{235}U is to be evaluated	Additional guidance may need to be developed to address transportation of high burnup and enriched TRISO fuel. The review method is limited to burnup credit available from actinide and fission product compositions associated with UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated in a pressurized-water reactor to an assembly-average burnup value not exceeding 60 GWd/MTU.
	Loading curve and burnup verification	Burnup credit loading curves; Performance of burnup verification	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.5.6 Reactor-Related Greater-Than-Class-C Waste and HLW (SL)		Not applicable because of irrelevant wastes	Not applicable	Not applicable	Not applicable

Table II-10. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent TRISO fuel				
Areas of review (NUREG-2215, Chapter 7)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.5.7 Supplemental Information	Not applicable	Not applicable	Not applicable	Not applicable

ANS. American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998 (R2007). “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.” La Grange Park, Illinois: American Nuclear Society. 2007.

DOE. “Safety Analysis Report for Fort St. Vrain Independent Spent Fuel Storage Installation.” Revision 8. Chapter 3. ML103640368. Idaho Falls, Idaho: U.S. Department of Energy Idaho Operations Office. 2010.

IAEA. “Advances in High Temperature Gas Cooled Reactor Fuel Technology.” IAEA-TECDOC-1674. Vienna, Austria: International Atomic Energy Agency. 2012.

_____. “Survey of Experience with Dry Storage of Spent Nuclear Fuel and Update of Wet Storage Experience.” Technical Report Series No. 290. STI/DOC/10/290. Vienna, Austria: International Atomic Energy Agency. 1988.

Jarrell, J. “A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium.” INL/EXT-18-51518. Idaho Falls, Idaho: Idaho National Laboratory. 2018.

NEA. “Technology Roadmap Update for Generation IV Nuclear Energy Systems.” OECD Nuclear Energy Agency. 2014.

NRC. “Safety Evaluation Report for License Renewal: Fort St. Vrain Independent Spent Fuel Storage Installation.” ML112000261. Washington, DC: U.S. Nuclear Regulatory Commission. 2011.

NRC. NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology.” Oak Ridge, Tennessee: Science Applications International Corporation. U.S. Nuclear Regulatory Commission. 2001.

_____. NUREG/CR-6361, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages.” ORNL/TM-13211. Oak Ridge, Tennessee: Oak Ridge National Laboratory. U.S. Nuclear Regulatory Commission. 1997.

NWTRB. “Management and Disposal of U.S. Department of Energy Spent Nuclear Fuel.” Washington, DC: Nuclear Waste Technical Review Board. 2017.

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
8.5.1 Drawings		Content of drawings	Information available on many NRC-certified storage container designs, particularly the design of the Fort St. Vrain ISFSI for storage of spent TRISO fuel (DOE, 2010; NRC, 2011); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.2 Codes and Standards	Usage and endorsement	Codes and standards used for the storage container design and construction	Codes and standards used to design the storage containers are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code case use and acceptability	Acceptability of ASME code cases	Specific code case referenced is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.3 Welding	Confinement weld design	Welding criteria and weld procedure qualification requirements	Detailed weld design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Confinement weld inspection	Weld inspection methods and requirements	Detailed weld inspection information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confinement weld testing	Weld testing methods and requirements	Detailed weld testing information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.4 Mechanical Properties of Metals	Tensile properties	Acceptability of material tensile properties	Mechanical properties for commonly used storage container materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fracture resistance	Acceptability of material fracture toughness	Mechanical properties for commonly used storage container materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Performance of aluminum components	Acceptability of the tensile properties, fracture toughness, and creep of aluminum materials	Mechanical properties for commonly used aluminum materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel

Areas of review (NUREG-2215, Chapter 8)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
8.5.5 Thermal Properties		Thermal properties of storage container materials; Effect of degradation and anisotropic dependencies of thermal properties	Thermal properties for commonly used storage container materials are available; Detailed evaluation of storage container components and fuels is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.6 Radiation Shielding Materials	Neutron shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Assessing previously unreviewed (new) neutron shielding materials	Temperature and radiation-induced degradation and its effects on neutron shield performance	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Gamma shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.7 Criticality Control Materials	Neutron absorbing (poison) material specification	Chemical composition, physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; Qualification testing	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Computation of percent credit for boron-based neutron absorbers	Level of credit allowed for absorber materials	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Qualifying the neutron absorber material fabrication process	Qualification of absorber material properties not associated with neutron attenuation	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.8 Concrete and Reinforcing Steel	Embedment materials	Material used for embedments, inserts, conduits, pipes, or other items embedded in the concrete	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Concrete design and temperature limits	Design and material specifications for the concrete; Temperature requirements; Changes in concrete properties with time	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Omission of reinforcement	Omission of reinforcing steel in the concrete	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Radiation damage	Radiation effects on concrete properties	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
8.5.9 Bolt Applications		Material properties of the bolting; Effect of corrosion on the bolting materials; Closure bolt stresses	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.10 Seals	Metallic seals	Material properties of metallic seals	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Elastomeric seals	Material properties of elastomeric seals; Effects of thermal, radiation, and chemical reactions on the seal materials	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.11 Corrosion Resistance	Environments	Range of environmental conditions encountered for storage container components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Carbon and low-alloy steels	Environment dependencies of corrosion rate; Coatings for corrosion prevention	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Austenitic stainless steels	Localized corrosion and stress corrosion cracking in chloride-containing environments; Chloride-induced stress corrosion cracking in sensitized stainless steels	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Duplex stainless steels	Microstructural alteration in welded duplex stainless steels; Fabrication and weld testing and acceptance criteria	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.12 Protective Coatings	Review guidance	Coating specifications	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Scope of coating application	Purpose of the coating, lists the components to be coated, and the expected environmental conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 8)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Coating selection	Coating manufacturer, type of primers and topcoat, coating thickness, and ability of the coating to withstand the in-service conditions	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating qualification testing	Qualification testing for coating performance	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
8.5.13 Content Reactions	Flammable and explosive reactions	Effects of flammable and explosive reactions among the content materials	Information available on the NRC-certified storage container design of the Fort St. Vrain ISFSI for storage of spent TRISO fuel (DOE, 2010; NRC, 2011); Detailed evaluation of storage container components and fuels is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel

Areas of review (NUREG-2215, Chapter 8)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Corrosion	Effects of corrosive reactions among the contents and between the contents and the storage container components	TRISO fuel discharged from fluoride salt-cooled high-temperature reactors (FHR) may contain residual salt coolant. Radiolysis of solid fluoride salts in radiation fields will generate fluorine gas that is toxic and potentially corrosive (Forsberg and Peterson, 2015).	Material performance of FHR fuel with residual salt material is to be evaluated	Additional guidance may need to be developed to address corrosion of non-fuel hardware associated with TRISO fuel. The SRP calls for examining whether corrosion wastage could lead to a loss of intended functions; however, for non-fuel hardware, the current review method is limited to guidance for the examination of corrosion of hardware components associated with stainless steel or zirconium alloy-clad UO_2 fuels.
8.5.14 Management of Aging Degradation	Initial storage term	Materials performance of storage container components for the duration of the storage term	Detailed evaluation of storage container materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Amendment applications submitted during a renewal	Aging management for the amendment applications	Detailed evaluation of storage container materials is expected	None expected	None identified; General acceptance criteria not impacted

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel

Areas of review (NUREG-2215, Chapter 8)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	review or after a renewal is issued		to be provided by the applicant		by specifics of technologies
8.5.15 Spent Fuel	Spent fuel classification	Classification of damaged, undamaged, and intact fuel	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Uncannistered spent fuel	Cladding alloys and maximum fuel burnup; Cladding mechanical properties; Effective cladding thickness; Maximum cladding temperature; Thermal cycling during loading operations; Composition of the cover gas; High burnup fuel monitoring and assessment; Release fractions	Although degradation of TRISO fuel has not been reported associated with dry storage conditions, information in the literature is limited except for the spent TRISO fuel stored at the Fort St. Vrain ISFSI (DOE, 2010, 1992; Marschman et al., 1993)	Performance of the coating layers on TRISO fuel under storage environments is to be evaluated	Additional guidance may need to be developed to address the mechanical properties of the coating layers for TRISO fuel. The current review method is limited to guidance for the examination of mechanical properties of zirconium alloy cladding, which is not applicable to TRISO fuel. It is necessary to examine whether similar or equivalent cladding functions are required in TRISO fuel.
	Cannistered spent fuel	Performance of the fuel can for damaged fuel	Detailed evaluation of the fuel can performance is expected to be	None expected	None identified; General acceptance criteria not impacted

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 8)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
			provided by the applicant		by specifics of technologies

DOE. “Safety Analysis Report for Fort St. Vrain Independent Spent Fuel Storage Installation.” Revision 8. Chapter 3. ML103640368. Idaho Falls, Idaho: U.S. Department of Energy Idaho Operations Office. 2010.

_____. “Characteristics of Potential Repository Wastes.” DOE/RW-0184-R1. Vol 2. ORNL/-2217, Vol 2. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1992.

Forsberg, C. and P.F. Peterson. “Spent Nuclear Fuel and Graphite Management for Salt-Cooled Reactors: Storage, Safeguards, and Repository Disposal.” Nuclear Technology. Vol. 191. pp. 113–121. 2015.

Marschman, S.C., F.M. Berting, R.G. Clemmer, E.R. Gilbert, R.J. Guenther, W.C. Morgan, and P. Sliva. “Characterization Plan for Fort St. Vrain and Peach Bottom Graphite Fuels.” PNNL-11365. Richland, Washington: Pacific Northwest National Laboratory. 1993.

NRC. “Safety Evaluation Report for License Renewal: Fort St. Vrain Independent Spent Fuel Storage Installation.” ML112000261. Washington, DC: U.S. Nuclear Regulatory Commission. 2011.

Table II-12. Information to be reviewed and potential gaps for evaluating confinement performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 9)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
9.5.1 Confinement Design Characteristics	Design criteria	Confinement design criteria	Information available on many NRC-certified storage container designs, particularly the design of the Fort St. Vrain ISFSI for storage of spent TRISO fuel (DOE, 2010; NRC, 2011); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Design features	Confinement design features	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
9.5.2 Confinement Monitoring Capability		Leakage test, monitoring systems, and surveillance requirements of the storage container	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
9.5.3 Nuclides with Potential for Release		Availability and release fractions of radioactive nuclides	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-12. Information to be reviewed and potential gaps for evaluating confinement performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 9)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
9.5.4 Confinement Analyses	Normal conditions	Confinement analysis and the resulting doses for the normal conditions	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Off-normal conditions (anticipated occurrences)	Confinement analysis and the resulting doses for the off-normal conditions	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Design-basis accident conditions (including natural phenomenon events)	Confinement analysis and the resulting doses for the accident conditions	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Identification of release events (SL)	Spectrum of release events for normal operations, off-normal operations, and design-basis accidents	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation of release estimates for spent nuclear fuel and high-level radioactive waste (SL)	Dose calculations and release estimates for normal operations, off-normal operations, and design-basis accidents	Detailed confinement evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table II-12. Information to be reviewed and potential gaps for evaluating confinement performance of dry storage systems and facilities for spent TRISO fuel					
Areas of review (NUREG-2215, Chapter 9)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Evaluation of release estimates for reactor-related greater than Class C waste (SL)	Not applicable to spent TRISO fuel	Not applicable	Not applicable	Not applicable
9.5.5 Supplemental Information		Not applicable	Not applicable	Not applicable	Not applicable
DOE. "Safety Analysis Report for Fort St. Vrain Independent Spent Fuel Storage Installation." Revision 8. Chapter 3. ML103640368. Idaho Falls, Idaho: U.S. Department of Energy Idaho Operations Office. 2010. NRC. "Safety Evaluation Report for License Renewal: Fort St. Vrain Independent Spent Fuel Storage Installation." ML112000261. Washington, DC: U.S. Nuclear Regulatory Commission. 2011.					

APPENDIX III

TRANSPORTATION OF SPENT FUEL

Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 2)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
2.4.1 Description of Structural Design	General	Drawings and descriptive information including weights and centers of gravity	Information available on many NRC-certified package designs (NRC, 2013); Detailed description of structural design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Identification of codes and standards for package design	Codes and standards used for the package design and fabrication	Codes and standards are available (NRC, 2013), but not specifically for fuel with nonsymmetrical contents (Oklo Inc., 2020)	Applicability of codes and standards for structural design of spent metal fuel is to be evaluated	None identified. The review method calls for verification of the code or standard developed for structures of similar design. Codes and standards to be used are expected to be defined or developed, or the technical basis will be provided for the adequacy of alternative codes and standards.
2.4.2 General Requirements for All Packages	Minimum package size	The smallest overall dimension of the package must not be less than 10 cm [4 in]	Specific package dimensions are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tamper-indicating feature	The package closure system must incorporate a tamper-indicating feature	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Positive closure	The package closure system must include a positive fastening device	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Package valve	A package valve or other device must be protected against unauthorized operation	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.3 Lifting and Tie-Down Standards for All Packages	Lifting devices	Lifting devices must be designed in accordance with 10 CFR 71.45(a)	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tie-down devices	Tie-down devices must be designed in accordance with 10 CFR 71.45(b)	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.4 General Considerations for Structural Evaluation of Packaging	Evaluation by analysis	Elements of the analysis used for structural evaluation	Specific analyses used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation by test	Elements of the test used for structural evaluation	Specific tests used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
2.4.5 Normal Conditions of Transport	Heat	Maximum temperature, maximum pressure, and thermal stress under the heat-loading condition	Structural performance of the package under the heat-loading condition is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Cold	Maximum temperature, minimum internal pressure, and residual stress under the cold condition	Structural performance of the package under the cold condition is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Reduced external pressure	Effects of reduced external pressure on the internal and external pressures of the package and the containment system	Effects of reduced external pressure is expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Increased external pressure	Effects of increased external pressure on the internal and external pressures of the package and the containment system	Effects of increased external pressure is expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Vibration and fatigue	Effects of vibration normally incident to transport; Fatigue under the combined stresses from vibration, temperature, and pressure loads	Information available on many NRC-certified package designs (NRC, 2013); however, no vibration test and analysis of packages for transporting sodium-containing spent metal fuel are available	Sodium creep and location shift susceptibility and its effects on the geometric configuration of spent metal fuel under the influence of vibration is to be evaluated	None identified. The review method calls for evaluating the package design for the effects of vibration. The existing review method is sufficient for the assessment of sodium creep under the influence of vibration.
	Water spray	Effects of the water spray test	Detailed water spray test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Free drop	Effects of the 0.3 to 1.2-m free-drop test	Information available on many NRC-certified package designs (NRC, 2013); however, no free-drop testing of packages for transporting sodium-containing spent metal fuel is available	Sodium creep and location shift susceptibility and its effects on the geometric configuration of spent nuclear metal fuel under the influence of drop is to be evaluated	None identified. The review method calls for evaluating the package design for the effects of free drop. The existing review method is sufficient for the assessment of sodium creep under the influence of a drop.
	Corner drop	Not applicable because of the package weight exceedance	Not applicable	Not applicable	Not applicable
	Compression	Not applicable because of the package weight exceedance	Not applicable	Not applicable	Not applicable

Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Penetration	Effects of the penetration test	Detailed penetration test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.6 Hypothetical Accident Conditions	Free drop	Effects of the 9-m free-drop test	Information available on many NRC-certified package designs (NRC, 2013); however, no 9-m free-drop testing of packages for spent metal fuel transportation is available	Ability of the metal fuel pins to withstand the specified drop conditions and maintain containment and criticality functions is to be evaluated	None identified. The review method calls for evaluating the package design for the effects of free drop. The existing review method is sufficient for the assessment of metal fuel integrity under the influence of drop.
	Crush	Not applicable because of the package weight exceedance	Not applicable	Not applicable	Not applicable
	Puncture	Effects of the puncture test	Detailed puncture test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Thermal	Effects of the fire test	Detailed fire test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 2)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Immersion	Effects of the immersion test	For some package designs, water may fill the package during the immersion test, thus applying hydrostatic pressure on the fuel rod and potentially compromising the cladding integrity	Ability of the cladding to withstand the increased external pressure from immersion test and its effects on fuel properties is to be evaluated	None identified. The review method calls for adequately evaluating the package design subjected to water pressure from immersion test. The existing review method is sufficient for the assessment of metal fuel integrity under the influence of water pressure from immersion test.
2.4.7 Air Transport Accident Conditions for Fissile Material		Not applicable because air transport is not anticipated	Not applicable	Not applicable	Not applicable
2.4.8 Special Requirement for Type B Packages Containing More Than 10^5 A_2		Applicability of 10 CFR 71.61 based on specific inventory estimate; Ability to withstand an external pressure of 2 MPa [290 psi] for 1 hour	Detailed analyses are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.9 Air Transport of Plutonium		Not applicable because air transport of plutonium is not anticipated	Not applicable	Not applicable	Not applicable

Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel				
Areas of review (NUREG-2216, Chapter 2)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
NRC. NUREG–0383, “Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance.” Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013. Oklo Inc. “Part II. Final Safety Analysis Report.” U.S. Nuclear Regulatory Commission ADAMS Accession Number ML20075A003. Sunnyvale, California: Oklo Inc. 2020.				

Table III-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
3.4.1 Description of the Thermal Design	Packaging design features	Drawings and description of the thermal features	Information available on many NRC-certified package designs (NRC, 2013); detailed description of thermal features is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used for the thermal design and evaluation of the package	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Content heat load specification	Maximum decay heat load; Methods and codes used to determine content decay heat loads	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Summary tables of temperatures	Maximum, minimum, and allowable temperatures of package components for normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 3)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Summary tables of pressures in the containment system	Design pressure limits of package components for normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.2 Material Properties and Component Specifications	Material thermal properties	Thermal properties of package materials; Sources of the thermal properties; Temperature-dependent thermal properties	Thermal properties for commonly used packaging materials are available; Some thermal properties of fuel pin components, structural components, and metal fuel are available (Leibowitz et al., 1976; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Accurate data to characterize phases, phase diagrams, heat capacity, and thermal properties of metal fuel are limited	None identified. The review method calls for verification of the thermal and thermomechanical properties, as well as their temperature dependence. The existing review method is sufficient to deal with the assessment of metal fuel properties important to the thermal analysis.
	Specifications of components	Maximum allowable service temperatures or pressures of package components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 3)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Thermal design limits of package materials and components	Maximum allowable temperatures of package components; Temperature limits of fuel and clad materials	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.3 General Considerations for Thermal Evaluations	Evaluation by analyses	Elements of the analysis used for thermal evaluation	Specific analyses used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation by Tests	Elements of the test used for thermal evaluation	Specific tests used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analysis	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Effects of uncertainties	Uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and analytical methods	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 3)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Conservatisms	Conservatisms associated with the thermal models and their effects on the safety parameters	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.4 Evaluation of Accessible Surface Temperatures		Thermal model used for calculating the accessible surface temperatures	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.5 Thermal Evaluation Under Normal Conditions of Transport	Heat and cold	Maximum accessible surface temperatures; Maximum temperatures of package components under the heat condition; Minimum temperatures of package components under the cold condition	The influence of heat and cold could lead to differential thermal expansion and stresses for the fuel pin components, thus potentially compromise the bonding between sodium and cladding and sodium and metal fuel slug	Thermal performance of the bonding between sodium and cladding and sodium and metal fuel slug under the heat and cold conditions is to be evaluated	None identified. The review method calls for examining that the tests for normal conditions of transport do not result in significant reduction in packaging effectiveness. The existing review method is sufficient to evaluate the performance of structure bonding and the metal fuel under the heat and cold conditions.

Table III-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Maximum normal operating pressure	The maximum normal operating pressure under the heat condition	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.6 Thermal Evaluation Under Hypothetical Accident Conditions	Initial conditions	Initial conditions of the package	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fire test	Effects of the fire test	For the hypothetical accident conditions, the temperature of the sodium inside the fuel pin may exceed the melting point and the resulting thermal stress may compromise cladding integrity	Ability of the metal fuel pins and the cladding to withstand the hypothetical accident conditions is to be evaluated	None identified. The review method calls for examining the evaluation of the package design regarding potential consequences of the fire test. The existing review method is sufficient to evaluate the integrity of metal fuel and cladding under the conditions consistent with the fire test.

Table III-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Maximum temperatures and pressures	The maximum temperatures and pressures of the package components under the hypothetical accident conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Janney, D.E. and S.L. Hayes. "Experimentally Known Properties of U-10Zr alloys: A critical Review." Nuclear Technology. Vol. 203. pp.109–128. 2018.
Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 1: Phases and Phase Diagrams." Nuclear Technology. Vol. 205. pp.1,387–1,415. 2019.
Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 2: Thermal and Mechanical Properties." Nuclear Technology. Vol. 206. pp.1–22. 2020.
Janney, D.E. "Metallic Fuels Handbook, Part 1: Alloys Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr, Including Those with Minor Actinides (Np, Am, Cm), Rare-earth Elements (La, Ce, Pr, Nd, Gd), and Y." INL/EXT-15-36520 Revision 3 Part 1. Idaho Falls, Idaho: Idaho National Laboratory. 2018a.
Janney, D.E. "Metallic Fuels Handbook, Part 2: Elements and Alloys not Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr." INL/EXT-15-36520 Revision 3 Part 2. Idaho Falls, Idaho: Idaho National Laboratory. 2018b.
Leibowitz, L., E.C. Chang, M.G. Chasanov, R.L. Gibby, C. Kim, A.C. Millunzi, D. Stahl. "Properties for Liquid Metal Fast Breeder Reactor Safety Analysis." Argonne National Laboratory. ANL-CEN-RSD-76-1. 1976.
NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table III-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 4)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
4.4.1 Description of the Containment System	Containment boundary	Containment design features including description of the containment boundary, containment boundary penetrations, method of closure, and leak test for penetrations.	Configuration of containment boundary varies depending on the package design for the specific contents (NRC, 2021, 2013); Detailed description of containment design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used for the containment design of the package	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Special requirements for damaged spent nuclear fuel	Not applicable to spent metal fuel	Not applicable	Not applicable	Not applicable
4.4.2 General Considerations	Type AF fissile packages	Not applicable to spent metal fuel	Not applicable	Not applicable	Not applicable

Table III-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
for Containment Evaluations	Type B packages	Contents and requirement for Type B packages	Spent metal fuel is expected to be transported using Type B packages; Detailed package design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Combustible-gas Generation	Combustible gases generated in the package do not exceed 5 percent by volume	Sodium reacts readily with water, which produces sodium hydroxide and hydrogen; the hydrogen can react violently with oxygen in air if ignited by a spark	Measures to ensure no failure of the containment boundary that would lead to quick reaction of sodium with inleakage water producing combustible gas are to be established	None identified. The review method identifies a 5 percent concentration threshold or lower if warranted by the flammable gas. In addition, Section 7.4.10.1 requires measures to remove moisture or oxygen from the container if metallic contents could potentially support pyrophoricity. The existing review method is sufficient to evaluate the effects of reactions of sodium and fuel with water and air in the context of transport of spent metal fuel.
4.4.3 Containment Evaluation under	Type B transportation packages	Releasable source term, maximum permissible	Information available on many NRC-certified package	Release rate calculations and criteria used to	None identified. The review method calls for ensuring the applicant

Table III-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
Normal Conditions of Transport		release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for normal conditions of transport in accordance with ANSI N14.5 (ANSI, 2014)	designs (NRC, 2021, 2013); Detailed containment evaluation is expected to be provided by the applicant	verify cladding welds and fuel pins integrity to ensure sufficient containment by the transportation package of the sodium-containing fuel under normal conditions of transport are to be evaluated	calculates the maximum permissible release rate and maximum permissible leakage rate in accordance with ANSI N14.5. The existing review method is sufficient to evaluate metal fuel integrity under normal conditions of transport.
	Spent nuclear fuel transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for normal conditions of transport in accordance with ANSI N14.5	Information available on many NRC-certified package designs (NRC, 2021, 2013); Detailed containment evaluation is expected to be provided by the applicant	Release rate calculations and criteria used to verify cladding welds and fuel pins integrity to ensure sufficient containment by the transportation package of the sodium-containing fuel under normal conditions of transport are to be evaluated	None identified. The review method calls for ensuring the applicant calculates the maximum permissible release rate and maximum permissible leakage rate in accordance with ANSI N14.5. The existing review method is sufficient to evaluate metal fuel integrity under normal conditions of transport.

Table III-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 4)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Compliance with containment design criteria	Packages must be designed to satisfy the containment requirements of 10 CFR 71.51(a)(1) under normal conditions of transport	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.4.4 Containment Evaluation Under Hypothetical Accident Conditions	Type B transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for hypothetical accident conditions in accordance with ANSI N14.5	Information available on many NRC-certified package designs (NRC, 2021, 2013); however, no data are available specifically for the metal fuel pin; Detailed containment evaluation is expected to be provided by the applicant	Release rate calculations of metal fuel cladding and welds from drop, fire, and other accident conditions to ensure sufficient containment by the transportation package of the sodium-containing fuel are to be evaluated	None identified. The review method calls for no escape of krypton-85 and other radioactive material, as well as external radiation dose rate not exceeding limits specified in 10 CFR 71.51(a)(2) for hypothetical accident conditions. The existing review method is sufficient to evaluate metal fuel integrity under hypothetical accident conditions.
	Spent nuclear fuel transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the	Information available on many NRC-certified package designs (NRC, 2021, 2013); however, no data are available specifically for the metal fuel pin;	Release rate calculations of metal fuel cladding and welds from drop, fire, and other accident conditions to ensure sufficient containment by the	None identified. The review method calls for no escape of krypton-85 and other radioactive material, as well as external radiation dose rate not exceeding limits

Table III-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		reference air leakage rate calculated for hypothetical accident conditions in accordance with ANSI N14.5	Detailed containment evaluation is expected to be provided by the applicant	transportation package of the sodium-containing fuel are to be evaluated	specified in 10 CFR 71.51(a)(2) for hypothetical accident conditions. The existing review method is sufficient to evaluate metal fuel integrity under hypothetical accident conditions.
	Compliance with containment design criteria	Packages must be designed to satisfy the containment requirements of 10 CFR 71.51(a)(2) under hypothetical accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

ANSI. ANSI N14.5-2014, "American National Standard for Radioactive Materials–Leakage Tests on Packages for Shipment." New York, New York: American National Standards Institute. 2014.

NRC. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9225 for the NAC-LWT Package." Revision 71. ML21078A200. Washington, DC: U.S. Nuclear Regulatory Commission. 2021.

_____. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table III-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.4.1 Description of the Shielding Design	Shielding design features	Drawings and description of the shielding design features	Information available on many NRC-certified package designs, particularly the NAC-LWT package for transport of metal fuel (NRC, 2021, 2013); Detailed shielding design is expected to be provided by the applicant	Shielding design for spent metal fuel with diverse sources of uranium is to be evaluated	None identified. The review method calls for evaluating a description of the shielding design features to ensure it addresses those items important to the package's shielding performance. The existing review method is sufficient to evaluate the shielding design dealing with spent metal fuel.
	Summary tables of maximum external radiation levels	Maximum radiation levels for all relevant package surfaces and appropriate distances from these surfaces under normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.2 Radioactive Materials and Source Terms	Source-term calculation methods	Methods used to determine the bounding source terms for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Gamma sources	Gamma source strengths and spectra for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Neutron sources	Neutron source strengths and spectra for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.3 Shielding Model and Model Specifications	Configuration of source and shielding	Dimensions and materials properties of the package contents, radioactive sources in the contents, and the packaging components	Information available on many NRC-certified package designs (NRC, 2021, 2013); Detailed shielding evaluation in a Type BF configuration is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Material properties (e.g., composition, mass densities, and atom densities) of packaging components, package contents, and the conveyance	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 5)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
5.4.4 Shielding Evaluation	Methods	Methods used for the shielding evaluations under normal and accident conditions	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code input and output data	Key input data and output files for the shielding evaluations	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fluence-rate-to-radiation-level conversion factors	Accuracy and acceptance of the conversion factors	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	External radiation levels	External radiation levels under normal and accident conditions	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

ASTM. C996. "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5 percent ^{235}U" <https://compass.astm.org/EDIT/html_annot.cgi?C996+20> (Accessed May 27, 2021). 2021.

NRC. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9225 for the NAC-LWT Package." Revision 71. ML21078A200. Washington, DC: U.S. Nuclear Regulatory Commission. 2021.

_____. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
6.4.1 Description of Criticality Design	Packaging design features	Design features important for criticality safety	Information available on many NRC-certified package designs (NRC, 2013); Detailed description of criticality features is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used in all aspects of the criticality design and evaluation	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Summary table of criticality evaluations	Maximum value of k_{eff}, uncertainty, bias and bias uncertainty for all relevant cases; Number of packages evaluated in the array cases	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Criticality safety index (CSI)	CSI limits for all package configurations	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.2 Fissile material contents		Content and type of fissile material	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
					specifics of technologies
6.4.3 General Considerations for Criticality Evaluations	Model configuration	Criticality evaluations demonstrating subcritical margins for single package and package arrays under normal and hypothetical conditions of transport	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Materials and their properties used in the criticality models	Information is expected to be available at the time of an application	None expected	None identified because this is a reporting of materials used for the criticality evaluation
	Analysis methods and nuclear data	Computer code and cross-section library used for criticality evaluations	Detailed criticality evaluation is expected to be provided by the applicant	None expected	None identified; The computer codes and cross section libraries are not impacted by specifics of technologies
	Demonstration of maximum reactivity	Analyses demonstrate the maximum k_{eff}	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Confirmatory analysis of the	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		criticality calculations			specifics of technologies
	Moderator exclusion under hypothetical accident conditions	Package subcriticality under hypothetical accident conditions	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.4 Single Package Evaluation	Configuration	Models for criticality evaluations confirming subcritical margins maintained for single package under normal and hypothetical accident conditions of transport	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Results	Results of the criticality calculations for single package	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.5 Evaluations of Package Arrays	Package arrays under normal conditions of transport	Criticality evaluation for an array of 5N packages that is subcritical under normal conditions of transport	Information available on many NRC-certified package designs (NRC, 2013). Although no prior experience for spent metal fuel package, information is expected	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
			to be available at the time of an application.		
	Evaluation of package arrays under hypothetical accident conditions	Criticality evaluation for an array of 2N packages that is subcritical under hypothetical accident conditions	Information available on many NRC-certified package designs (NRC, 2013). Although no prior experience for spent metal fuel package, information is expected to be available at the time of an application.	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Package arrays results and criticality safety index	Appropriate N value is used to ascertain the CSI	Information available on many NRC-certified package designs (NRC, 2013). Although no prior experience for spent metal fuel package, information is expected to be available at the time of an application.	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
6.4.6 Benchmark Evaluations	Experiments and applicability	Benchmarking computer codes for criticality calculations against fitting critical experiments	Information available on many NRC-certified package designs (NRC, 2013); however, no information for sodium-containing spent metal fuel was found. Criticality benchmark data and validation of existing criticality codes and methods are limited for initial enrichments between 5 and 20 weight percent (Jarrell, 2018).	Criticality benchmark data and applicability of existing criticality codes and methods for spent metal fuel with initial enrichments between 5 and 20 weight percent ^{235}U are to be evaluated	None identified. The review method calls for verifying the applicant has benchmarked the computer codes used for criticality calculations against appropriate critical experiments applicable to the actual packaging design and contents. The existing review method is sufficient to deal with the availability of criticality benchmark data and applicability of existing criticality codes and methods for spent metal fuel.

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Bias determination	Results of the benchmark calculations and bias evaluations	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001). No criticality benchmarking for spent metal fuel with initial higher enrichment was found.	Criticality benchmarking for spent metal fuel with higher enrichment is to be evaluated, given the potential lack of criticality benchmark data	None identified. The review method calls for evaluating whether the applicant demonstrates that the benchmark calculations are adequately converged and justifies the bias and bias uncertainty. The existing review method is sufficient to deal with criticality benchmarking for spent metal fuel.
6.4.7 Burnup Credit Evaluation for Commercial Light-Water Reactor Spent Nuclear Fuel	Limits for the certification basis	Analytic methods, assumptions, and assay date used in the burnup credit analyses for the certification basis	Metal fuel has an enrichment of 26–93 weight percent ^{235}U and a fuel burnup of 38–143 GWd/MTU (FRWG, 2018)	Burnup credit analyses for spent metal fuel package designs with increasing enrichment and fuel burnup limits are to be evaluated, given the potential lack of code validation data	Additional guidance may need to be developed to address transportation of high burnup and enriched metal fuel. The review method specifies the current licensed fuel burnup and enrichment limits for transporting light water reactor fuel (i.e., 60 GWd/MTU burnup and 5.0 weight percent ^{235}U enrichment).

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Model assumptions	Models and analysis assumptions for the k_{eff} calculations representative of the physics in the package	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code validation—isotopic depletion	Validation of the depletion codes; Bias and bias uncertainty of the codes	Metal fuel has an enrichment up to 20 weight percent ^{235}U and a fuel burnup of 38–143 GWd/MTU (FRWG, 2018)	Depletion analyses for spent metal fuel package designs with increasing enrichment and fuel burnup limits are to be evaluated, given the potential lack of code validation data	Additional guidance may need to be developed to address transportation of high burnup and enriched metal fuel. The review method is limited to burnup credit available from actinide compositions associated with UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated in a pressurized-water reactor to an assembly-average burnup value not exceeding 60 GWd/MTU.

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Code validation—k_{eff} determination	Bias and bias uncertainty associated with actinide-only, and fission product and minor actinide burnup credit	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001); however, no criticality benchmarking for spent metal fuel with higher initial enrichments was found.	Criticality benchmarking for spent metal fuel with initial enrichments between 5 and 20 weight percent ^{235}U is to be evaluated	Additional guidance may need to be developed to address transportation of high burnup and enriched metal fuel. The review method is limited to burnup credit available from actinide and fission product compositions associated with UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated in a pressurized-water reactor to an assembly-average burnup value not exceeding 60 GWd/MTU.
	Loading curve and burnup verification	Burnup credit loading curves; Performance of burnup verification	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent metal fuel				
Areas of review (NUREG-2216, Chapter 6)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**

ANS. American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998 (R2007). "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." La Grange Park, Illinois: American Nuclear Society. 2007.

Jarrell, J. "A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium." INL/EXT-18-51518. Idaho Falls, Idaho: Idaho National Laboratory. 2018.

NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

_____. NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology." Oak Ridge, Tennessee: Science Applications International Corporation. U.S. Nuclear Regulatory Commission. 2001.

_____. NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages." ORNL/TM-11936. Oak Ridge, TN: Oak Ridge National Laboratory. U.S. Nuclear Regulatory Commission. 1997.

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.1 Drawings		Content of drawings	Information available on many NRC-certified package designs (NRC, 2013); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for examining the content of engineering drawings as well as the description of materials in package designs
7.4.2 Codes and Standards	Usage and endorsement	Codes and standards used for the package design and construction	Codes and standards are available (NRC, 2013); however, it is uncertain whether those standards would apply to new materials potentially to be used for package design and fabrication for transport of metal fuel	Applicability of codes and standards for package design and fabrication with new materials is to be evaluated	None identified. The review method calls for verification of the codes and standards for packaging components important to safety. Codes and standards to be used are expected to be defined or developed, or the technical basis will be provided for the adequacy of alternative codes and standards.
	ASME code components	Construction of ASME code components	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code case use/acceptability	Acceptability of ASME code cases	Specific code case referenced is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Non-ASME code components	Construction of non-ASME code components	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.3 Weld Design and Inspection	Weld Design and Inspection	Welding criteria and weld procedure qualification requirements	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that standard welding processes are adequate for package design and fabrication for transport of metal fuel. If new technologies were used in the design and fabrication of welds, the SRP calls for examination of compliance with any established codes and standards proposed in the application on design and construction.
	Moderator exclusion for commercial spent nuclear fuel packages under hypothetical accident conditions	Not applicable to packages for spent metal fuel transportation	Not applicable	Not applicable	Not applicable

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.4.4 Mechanical Properties	Tensile properties	Acceptability of material tensile properties	Mechanical properties for commonly used packaging materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that commonly used packaging materials may be also adequate for package design and fabrication for transport of metal fuel. If alternative or new materials were required in the design and fabrication of transportation packages, the SRP calls for examination of the adequacy of information in the application related to mechanical properties of those alternative materials.
	Fracture resistance	Acceptability of material fracture toughness	Mechanical properties for commonly used packaging materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tensile properties and creep of aluminum alloys at elevated temperatures	Acceptability of the tensile properties and creep of aluminum alloys	Mechanical properties for commonly used aluminum alloys are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Impact limiters	Acceptability of the mechanical properties of the impact limiter materials	Mechanical properties for commonly used impact limiter materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.5 Thermal Properties of Materials		Thermal properties of package materials; Effect of degradation and anisotropic dependencies of thermal properties	Thermal properties for commonly used packaging materials are available; Some thermal properties of fuel pin components, structural components, and metal fuel are available (Leibowitz et al., 1976; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Data characterizing phases, phase diagrams, heat capacity, and thermal properties of metal fuel and converted waste forms are limited	None identified. The review method calls for verification of the thermal properties and the change in these properties from material degradation. The existing review method is sufficient to deal with the availability of information related to metal fuel properties important to the thermal analysis.
7.4.6 Radiation Shielding	Neutron-shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Gamma-shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.7 Criticality Control	Neutron-absorbing (poison) material specification	Chemical composition, physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; Qualification testing	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Computation of percent credit for boron-based neutron absorbers	Level of credit allowed for absorber materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Qualifying properties not associated with attenuation	Qualification of absorber material properties not associated with neutron attenuation	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.8 Corrosion Resistance	Environments	Range of environmental conditions encountered for package components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Carbon and low-alloy steels	Environment dependencies of corrosion rate; Coatings for corrosion prevention	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Austenitic stainless steel	Localized corrosion and chloride-induced stress corrosion cracking in chloride-containing environments; Intergranular corrosion and stress corrosion cracking in sensitized stainless steel	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.4.9 Protective Coatings	Review guidance	Coating specifications	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Scope of coating application	Purpose of the coating, lists the components to be coated, and the expected environmental conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating selection	Coating manufacturer, type of primers and topcoat, coating thickness, and ability of the coating to withstand the in-service conditions	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating qualification testing	Qualification testing for coating performance in accordance with several standard ASTM (and possibly other) tests	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for evaluating any qualification testing for the demonstration of coating performance.

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.10 Content Reactions	Flammable and explosive reactions	Effects of flammable and explosive reactions among the content materials	Sodium reacts violently with water, which produces sodium hydroxide and hydrogen, and the hydrogen burns when in contact with air	Safety protocols in transporting sodium-containing metal fuel are to be established	None identified. The review method calls for measures to remove moisture or oxygen to be demonstrated. The existing review method is sufficient to evaluate the effects of reactions of sodium and fuel with water in the context of transport of spent metal fuel.
	Content chemical reactions, outgassing, and corrosion	Effects of chemical reactions, outgassing, and corrosion among the contents and between the contents and the package components	Some data are available on metal fuel (Carmack et al., 2009; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Effects of air and water on chemical interaction and galvanic coupling of package internal materials including sodium in the fuel pin are to be evaluated	Additional guidance may need to be developed to address the corrosion of non-fuel hardware associated with metal fuel. The SRP calls for examining that corrosion wastage will not lead to a loss of intended functions; however, for non-fuel hardware the current review method is limited to guidance for the examination of corrosion of hardware components associated with stainless steel or zirconium alloy-clad UO_2 fuels.

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel

Areas of review (NUREG-2216, Chapter 7)	Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.4.11 Radiation Effects	Effects of radiation on the performance of the package materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. Commonly used packaging materials may be also adequate for package design and fabrication for transport of metal fuel. If alternative or new materials were required in the design and fabrication of transportation packages, the SRP calls for examination of the adequacy of information in the application related to radiation effects on those alternative materials.

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.12 Package Contents		Chemical and physical form of the package contents; Effects of corrosion, chemical reactions, and radiation on the properties of the contents	Some data are available on metal fuel (Carmack et al., 2009; Janney, 2018a,b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Effects of air and water on chemical interaction and galvanic coupling of package internal materials including sodium in the fuel pin are to be evaluated	None identified. The review method calls for evaluating effects of corrosion, chemical reactions, and radiation. The existing review method is sufficient to evaluate the effects of air and water on chemical interaction and galvanic coupling of the package for transporting spent metal fuel.
7.4.13 Fresh (Unirradiated) Fuel Cladding		Not applicable to spent metal fuel	Not applicable	Not applicable	Not applicable
7.4.14 Spent Nuclear Fuel	Spent fuel classification	Classification of damaged, undamaged, and intact fuel	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Uncannistered spent fuel	Cladding alloys and maximum fuel burnup; Cladding mechanical properties; Effective cladding thickness; Maximum cladding temperature; Thermal cycling during loading operations; Composition of the cover gas; High burnup fuel monitoring and assessment; Release fractions	Stainless steel cladding performance in transportation packages may be challenged by sensitization, intergranular attack, stress corrosion cracking, thermal aging, and radiation embrittlement (Alexander and Nanstand, 1995; Chandra et al., 2012; Guenther et al., 1996). Spent metal fuel may experience degradation during transportation, particularly oxidation, hydriding, fragmentation, and restructuring-swelling (Guenther et al., 1996).	Performance of stainless steel cladding and spent metal fuel under transportation environments is to be evaluated; Advanced cladding material properties that can be used to achieve high burnup are to be evaluated, especially material performance data under the influence of irradiation	Additional guidance may need to be developed to address mechanical properties of stainless steel and advanced cladding materials for metal fuel. The current review method is limited to guidance for the performance of zirconium alloy, aluminum alloy, or stainless steel-clad UO_2 fuels.
	Cannistered spent fuel	Performance of the fuel can for damaged fuel	Detailed evaluation of the fuel can performance is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.15 Bolting Material		Material properties of the bolting; Effects of corrosion, chemical reactions, and radiation on the bolting materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.16 Seals	Metallic seals	Material properties of metallic seals; Effects of corrosion, chemical reactions, and radiation on the seal materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Elastomeric seals	Material properties of elastomeric seals; Effects of corrosion, chemical reactions, and radiation on the seal materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel				
Areas of review (NUREG-2216, Chapter 7)	Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
Carmack, W., D. Porter, Y.H.S. Chang, M. Meyer, D. Burkes, C. Lee, T. Mizuno, F. Delage, and J. Somers. "Metallic Fuels for Advanced Reactors." *Journal of Nuclear Materials.* Vol. 392. pp. 139–150. 2009. Garner, F.A. "Irradiation Performance of Cladding and Structural Steels in Liquid Metal Reactors." Nuclear Materials: Part 1. Materials Science and Technology: A Comprehensive Treatment. Frost, B.R.T., Editor. VCH Publishers. pp. 419–543. 1993. Guenther, R.J., A.B. Johnson, A.L. Lund, E.R. Gilbert, S.P. Pednekar, F.M. Berting, L.L. Burger, S.A. Bryan, and T.M. Orlando. "Initial Evaluation of Dry Storage Issues for Spent Nuclear Fuels in Wet Storage at the Idaho Chemical Processing Plant." INEL-96/0140. Idaho National Engineering Laboratory. 1996. IAEA. "Structural Materials for Liquid Metal Cooled Fast Reactor Fuel Assemblies: Operational Behaviour." Nuclear Energy Series No. NF-T-4.3. Vienna, Austria: International Atomic Energy Agency. 2012. Janney, D.E. "Metallic Fuels Handbook, Part 1: Alloys Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr, Including Those with Minor Actinides (Np, Am, Cm), Rare-earth Elements (La, Ce, Pr, Nd, Gd), and Y." INL/EXT-15-36520 Revision 3 Part 1. Idaho Falls, Idaho: Idaho National Laboratory. 2018a. Janney, D.E. "Metallic Fuels Handbook, Part 2: Elements and Alloys not Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr." INL/EXT-15-36520 Revision 3 Part 2. Idaho Falls, Idaho: Idaho National Laboratory. 2018b. Janney, D.E. and S.L. Hayes. "Experimentally Known Properties of U-10Zr alloys: A critical Review." *Nuclear Technology*. Vol. 203. pp.109–128. 2018. Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 1: Phases and Phase Diagrams." *Nuclear Technology*. Vol. 205. pp.1,387–1,415. 2019. Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 2: Thermal and Mechanical Properties." *Nuclear Technology*. Vol. 206. pp.1–22. 2020. NRC. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.				

Table III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 2)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
2.4.1 Description of Structural Design	General	Drawings and descriptive information including weights and centers of gravity	Information available on many NRC-certified package designs, particularly the TN-FSV and NAC-LWT packages for transport of TRISO fuel (NRC, 2021, 2014, 2013); Detailed description of structural design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Identification of codes and standards for package design	Codes and standards used for the package design and fabrication	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.2 General Requirements for All Packages	Minimum package size	The smallest overall dimension of the package must not be less than 10 cm [4 in]	Specific package dimensions are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tamper-indicating feature	The package closure system must incorporate a tamper-indicating feature	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 2)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Positive closure	The package closure system must include a positive fastening device	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Package valve	A package valve or other device must be protected against unauthorized operation	Detailed package design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.3 Lifting and Tie-Down Standards for All Packages	Lifting devices	Lifting devices must be designed in accordance with 10 CFR 71.45(a)	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tie-down devices	Tie-down devices must be designed in accordance with 10 CFR 71.45(b)	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.4 General Considerations for Structural Evaluation of Packaging	Evaluation by analysis	Elements of the analysis used for structural evaluation	Specific analyses used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation by test	Elements of the test used for structural evaluation	Specific tests used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
2.4.5 Normal Conditions of Transport	Heat	Maximum temperature, maximum pressure, and thermal stress under the heat-loading condition	Structural performance of the package under the heat-loading condition is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Cold	Maximum temperature, minimum internal pressure, and residual stress under the cold condition	Structural performance of the package under the cold condition is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Reduced external pressure	Effects of reduced external pressure on the internal and external pressures of the package and the containment system	Effects of reduced external pressure is expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Increased external pressure	Effects of increased external pressure on the internal and external pressures of the package and the containment system	Effects of increased external pressure is expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 2)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Vibration and fatigue	Effects of vibration normally incident to transport; Fatigue under the combined stresses from vibration, temperature, and pressure loads	Effects of vibration and fatigue analysis are expected to be described and evaluated by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Water spray	Effects of the water spray test	Detailed water spray test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Free drop	Effects of the 0.3 to 1.2-m free-drop test	Detailed free-drop test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Corner drop	Not applicable because of the package weight exceedance	Not applicable	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Compression	Not applicable because of the package weight exceedance	Not applicable	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 2)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Penetration	Effects of the penetration test	Detailed penetration test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.6 Hypothetical Accident Conditions	Free drop	Effects of the 9-m free-drop test	Detailed free-drop test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Crush	Not applicable because of the package weight exceedance	Not applicable	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Puncture	Effects of the puncture test	Detailed puncture test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Thermal	Effects of the fire test	Detailed fire test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Immersion	Effects of the immersion test	Detailed immersion test is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent TRISO fuel				
Areas of review (NUREG-2216, Chapter 2)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
2.4.7 Air Transport Accident Conditions for Fissile Material	Not applicable because air transport is not anticipated	Not applicable	Not applicable	Not applicable
2.4.8 Special Requirement for Type B Packages Containing More Than 10^5 A_2	Applicability of 10 CFR 71.61 based on specific inventory estimate; Ability to withstand an external pressure of 2 MPa [290 psi] for 1 hour	Detailed analyses are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
2.4.9 Air Transport of Plutonium	Not applicable because air transport of plutonium is not anticipated	Not applicable	Not applicable	Not applicable

NRC. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9225 for the NAC-LWT Package." Revision 71. ML21078A200. Washington, DC: U.S. Nuclear Regulatory Commission. 2021.

_____. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9253 for the TN-FSV Package." Revision 13. ML14167A316. Washington, DC: U.S. Nuclear Regulatory Commission. 2014.

_____. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table III-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
3.4.1 Description of the Thermal Design	Packaging design features	Drawings and description of the thermal features	Information available on many NRC-certified package designs, particularly the TN-FSV and NAC-LWT packages for transport of TRISO fuel (NRC, 2021, 2014, 2013); Detailed description of thermal features is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used for the thermal design and evaluation of the package	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Content heat load specification	Maximum decay heat load; Methods and codes used to determine content decay heat loads	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Summary tables of temperatures	Maximum, minimum, and allowable temperatures of package components for normal and	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
		accident conditions			
	Summary tables of pressures in the containment system	Design pressure limits of package components for normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.2 Material Properties and Component Specifications	Material thermal properties	Thermal properties of package materials; Sources of the thermal properties; Temperature-dependent thermal properties	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Specifications of components	Maximum allowable service temperatures or pressures of package components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Thermal design limits of package materials and components	Maximum allowable temperatures of package components; Temperature limits of fuel and clad materials	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
3.4.3 General Considerations for Thermal Evaluations	Evaluation by analyses	Elements of the analysis used for thermal evaluation	Specific analyses used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation by Tests	Elements of the test used for thermal evaluation	Specific tests used to evaluate the package are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analysis	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Effects of uncertainties	Uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and analytical methods	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Conservatisms	Conservatisms associated with the thermal models and their effects on the safety parameters	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 3)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
3.4.4 Evaluation of Accessible Surface Temperatures		Thermal model used for calculating the accessible surface temperatures	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.5 Thermal Evaluation Under Normal Conditions of Transport	Heat and cold	Maximum accessible surface temperatures; Maximum temperatures of package components under the heat condition; Minimum temperatures of package components under the cold condition	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Maximum normal operating pressure	The maximum normal operating pressure under the heat condition	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
3.4.6 Thermal Evaluation Under Hypothetical Accident Conditions	Initial conditions	Initial conditions of the package	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 3)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Fire test	Effects of the fire test	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Maximum temperatures and pressures	The maximum temperatures and pressures of the package components under the hypothetical accident conditions	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

NRC. “Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9225 for the NAC-LWT Package.” Revision 71. ML21078A200. Washington, DC: U.S. Nuclear Regulatory Commission. 2021.

_____. “Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9253 for the TN-FSV Package.” Revision 13. ML14167A316. Washington, DC: U.S. Nuclear Regulatory Commission. 2014.

_____. NUREG–0383, “Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance.” Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table III-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
4.4.1 Description of the Containment System	Containment boundary	Containment design features including description of the containment boundary, containment boundary penetrations, method of closure, and leak test for penetrations.	Information available on many NRC-certified package designs, particularly the TN-FSV and NAC-LWT packages for transport of TRISO fuel (NRC, 2021, 2014, 2013); Detailed description of containment design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used for the containment design of the package	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Special requirements for damaged spent nuclear fuel	Not applicable to spent TRISO fuel	Not applicable	Not applicable	Not applicable
4.4.2 General Considerations for Containment Evaluations	Type AF fissile packages	Not applicable to spent TRISO fuel	Not applicable	Not applicable	Not applicable
	Type B packages	Contents and requirement for Type B packages	Spent TRISO fuel is expected to be transported using Type B packages; Detailed package design is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Combustible-gas Generation	Combustible gases generated in the package do not exceed 5 percent by volume	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
4.4.3 Containment Evaluation under Normal Conditions of Transport	Type B transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for normal conditions of transport in accordance with ANSI N14.5 (ANSI, 2014)	Existing NRC-certified TN-FSV and NAC-LWT packages (NRC, 2021, 2014) provide all applicable containment evaluation under the normal conditions of transport; Detailed package containment evaluations under normal conditions of transport are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 4)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Spent nuclear fuel transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for normal conditions of transport in accordance with ANSI N14.5	Existing NRC-certified TN-FSV and NAC-LWT packages (NRC, 2021, 2014) provide all applicable containment evaluation under the normal conditions of transport; Detailed package containment evaluations under normal conditions of transport are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Compliance with containment design criteria	Packages must be designed to satisfy the containment requirements of 10 CFR 71.51(a)(1) under normal conditions of transport	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 4)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
4.4.4 Containment Evaluation Under Hypothetical Accident Conditions	Type B transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for hypothetical accident conditions in accordance with ANSI N14.5	Existing NRC-certified TN-FSV and NAC-LWT packages (NRC, 2021, 2014) provide applicable containment evaluation under hypothetical accident conditions; Detailed package containment evaluation under hypothetical accident conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Spent nuclear fuel transportation packages	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for hypothetical accident conditions in accordance with ANSI N14.5	Existing NRC-certified TN-FSV and NAC-LWT packages (NRC, 2021, 2014) provide applicable containment evaluation under hypothetical accident conditions; Detailed package containment evaluation under hypothetical accident conditions is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 4)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Compliance with containment design criteria	Packages must be designed to satisfy the containment requirements of 10 CFR 71.51(a)(2) under hypothetical accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

ANSI. ANSI N14.5-2014, "American National Standard for Radioactive Materials–Leakage Tests on Packages for Shipment." New York, New York: American National Standards Institute. 2014.

NRC. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9225 for the NAC-LWT Package." Revision 71. ML21078A200. Washington, DC: U.S. Nuclear Regulatory Commission. 2021.

_____. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9253 for the TN-FSV Package." Revision 13. ML14167A316. Washington, DC: U.S. Nuclear Regulatory Commission. 2014.

_____. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table III-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 5)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
5.4.1 Description of the Shielding Design	Shielding design features	Drawings and description of the shielding design features	Information available on many NRC-certified package designs, particularly the TN-FSV and NAC-LWT packages for transport of TRISO fuel (NRC, 2021, 2014, 2013); Detailed shielding design is expected to be provided by the applicant	Shielding design for spent TRISO fuel with diverse sources of uranium is to be evaluated	None identified. The review method calls for evaluating a description of the shielding design features to ensure it addresses those items important to the package's shielding performance. The existing review method is sufficient to support the evaluation of shielding performance of spent TRISO fuel.
	Summary tables of maximum external radiation levels	Maximum radiation levels for all relevant package surfaces and appropriate distances from these surfaces under normal and accident conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.2 Radioactive Materials and Source Terms	Source-term calculation methods	Methods used to determine the bounding source terms for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 5)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Gamma sources	Gamma source strengths and spectra for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Neutron sources	Neutron source strengths and spectra for the package contents	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
5.4.3 Shielding Model and Model Specifications	Configuration of source and shielding	Dimensions and materials properties of the package contents, radioactive sources in the contents, and the packaging components	Information available on many NRC-certified package designs (NRC, 2021, 2014, 2013); Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Material properties (e.g., composition, mass densities, and atom densities) of packaging components, package contents, and the conveyance	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 5)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
5.4.4 Shielding Evaluation	Methods	Methods used for the shielding evaluations under normal and accident conditions	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code input and output data	Key input data and output files for the shielding evaluations	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Fluence-rate-to-radiation-level conversion factors	Accuracy and acceptance of the conversion factors	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	External radiation levels	External radiation levels under normal and accident conditions	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Rigor of the confirmatory analyses	Detailed shielding evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

ASTM. C996. "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5 percent ^{235}U" <https://compass.astm.org/EDIT/html_annot.cgi?C996+20> (Accessed May 27, 2021). 2021.

NRC. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9225 for the NAC-LWT Package." Revision 71. ML21078A200. Washington, DC: U.S. Nuclear Regulatory Commission. 2021.

______. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9253 for the TN-FSV Package." Revision 13. ML14167A316. Washington, DC: U.S. Nuclear Regulatory Commission. 2014.

______. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
6.4.1 Description of Criticality Design	Packaging design features	Design features important for criticality safety	Information available on many NRC-certified package designs, particularly the TN-FSV and NAC-LWT packages for transport of TRISO fuel (NRC, 2021, 2014, 2013); Detailed description of criticality features is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Codes and standards	Codes and standards used in all aspects of the criticality design and evaluation	Codes and standards used to design the package are expected to be available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Summary table of criticality evaluations	Maximum value of k_{eff}, uncertainty, bias and bias uncertainty for all relevant cases; Number of packages evaluated in the array cases	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Criticality safety index (CSI)	CSI limits for all package configurations	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.2 Fissile material contents		Content and type of fissile material	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.3 General Considerations for Criticality Evaluations	Model configuration	Criticality evaluations demonstrating subcritical margins for single package and package arrays under normal and hypothetical conditions of transport	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Material properties	Materials and their properties used in the criticality models	Information is expected to be available at the time of an application	None expected	None identified, because this is a reporting of materials used for the criticality evaluation
	Analysis methods and nuclear data	Computer code and cross-section library used for criticality evaluations	Detailed criticality evaluation is expected to be provided by the applicant	None expected	None identified; The computer codes and cross section libraries are not impacted by specifics of technologies

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Demonstration of maximum reactivity	Analyses demonstrate the maximum k_{eff}	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Confirmatory analyses	Confirmatory analysis of the criticality calculations	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Moderator exclusion under hypothetical accident conditions	Package subcriticality under hypothetical accident conditions	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
6.4.4 Single Package Evaluation	Configuration	Models for criticality evaluations confirming subcritical margins maintained for single package under normal and hypothetical accident conditions of transport	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Results	Results of the criticality calculations for single package	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
					specifics of technologies
6.4.5 Evaluations of Package Arrays	Package arrays under normal conditions of transport	Criticality evaluation for an array of 5N packages that is subcritical under normal conditions of transport	Information available on many NRC-certified package designs, particularly the TN-FSV and NAC-LWT packages for transport of spent TRISO fuel (NRC, 2021, 2014, 2013); Detailed criticality evaluations are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Evaluation of package arrays under hypothetical accident conditions	Criticality evaluation for an array of 2N packages that is subcritical under hypothetical accident conditions	Detailed criticality evaluations are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Package arrays results and criticality safety index	Appropriate N value is used to ascertain the CSI	Detailed criticality evaluations are expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
6.4.6 Benchmark Evaluations	Experiments and applicability	Benchmarking computer codes for criticality calculations against fitting critical experiments	Information available on many NRC-certified package designs, particularly the TN-FSV and NAC-LWT packages for transport of spent TRISO fuel (NRC, 2021, 2014, 2013); however, criticality benchmark data and validation of existing criticality codes and methods are limited for enrichments between 5 and 20 weight percent (Jarrell, 2018)	Criticality benchmark data and applicability of existing criticality codes and methods for spent TRISO fuel with initial enrichments between 5 and 20 weight percent ^{235}U are to be evaluated	None identified. The review method calls for verifying the applicant has benchmarked the computer codes used for criticality calculations against appropriate critical experiments applicable to the actual packaging design and contents. The existing review method is sufficient to deal with the availability of criticality benchmark data and applicability of existing criticality codes and methods for spent TRISO fuel.

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 6)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Bias determination	Results of the benchmark calculations and bias evaluations	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001). Criticality benchmarking for spent TRISO fuel with higher enrichment is limited.	Criticality benchmarking for spent TRISO fuel with higher enrichment is to be evaluated, given the potential lack of criticality benchmark data	None identified. The review method calls for evaluating whether the applicant demonstrates that the benchmark calculations are adequately converged and justifies the bias and bias uncertainty. The existing review method is sufficient to deal with criticality benchmarking for spent TRISO fuel.
6.4.7 Burnup Credit Evaluation for Commercial Light-Water Reactor Spent Nuclear Fuel	Limits for the certification basis	Analytic methods, assumptions, and assay date used in the burnup credit analyses for the certification basis	TRISO fuel has an enrichment up to 20 weight percent ^{235}U and a fuel burnup of 150–200 GWd/MTU (NEA, 2014)	Burnup credit analyses for spent TRISO fuel package designs with increasing enrichment and fuel burnup limits are to be evaluated, given the potential lack of code validation data	Additional guidance may need to be developed to address transportation of high burnup and enriched TRISO fuel. The review method specifies the current licensed fuel burnup and enrichment limits for transporting light water reactor fuel (i.e., 60 GWd/MTU burnup and 5.0 weight percent ^{235}U initial enrichment).

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Model assumptions	Models and analysis assumptions for the k_{eff} calculations representative of the physics in the package	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Code validation—isotopic depletion	Validation of the depletion codes; Bias and bias uncertainty of the codes	TRISO fuel has an enrichment up to 20 weight percent ^{235}U and a fuel burnup of 150–200 GWd/MTU (NEA, 2014)	Depletion analyses for spent TRISO fuel package designs with increasing enrichment and fuel burnup limits are to be evaluated, given the potential lack of code validation data	Additional guidance may need to be developed to address transportation of high burnup and enriched TRISO fuel. The review method is limited to burnup credit available from actinide compositions associated with UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated in a pressurized-water reactor to an assembly-average burnup value not exceeding 60 GWd/MTU.

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 6)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Code validation—k_{eff} determination	Bias and bias uncertainty associated with actinide-only, and fission product and minor actinide burnup credit	There are several guidance documents on benchmarking criticality evaluations (ANS, 2007; NRC, 1997, 2001); however, no criticality benchmarking for spent TRISO fuel with higher enrichments was found.	Criticality benchmarking for spent TRISO fuel with enrichments between 5 and 20 weight percent ^{235}U is to be evaluated	Additional guidance may need to be developed to address transportation of high burnup and enriched TRISO fuel. The review method is limited to burnup credit available from actinide and fission product compositions associated with UO_2 fuel enriched up to 5.0 weight percent ^{235}U that has been irradiated in a pressurized-water reactor to an assembly-average burnup value not exceeding 60 GWd/MTU.
	Loading curve and burnup verification	Burnup credit loading curves; Performance of burnup verification	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

<table>
<tr><th colspan="5">Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel</th></tr>
<tr><th>Areas of review (NUREG-2216, Chapter 6)</th><th>Key information to be reviewed</th><th>Information availability</th><th>Potential information needs</th><th>Potential guidance gaps</th></tr>
<tr><td colspan="5">ANS. American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1-1998 (R2007). “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.” La Grange Park, Illinois: American Nuclear Society. 2007.
Jarrell, J. “A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium.” INL/EXT-18-51518. Idaho Falls, Idaho: Idaho National Laboratory. 2018.
NRC. “Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9225 for the NAC-LWT Package.” Revision 71. ML21078A200. Washington, DC: U.S. Nuclear Regulatory Commission. 2021.
_____. “Safety Evaluation Report for Model No. Versa-Pac Package Certificate of Compliance No. 9342 Revision No. 15.” ML20139A037. Washington, DC: U.S. Nuclear Regulatory Commission. 2020.
_____. “Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9253 for the TN-FSV Package.” Revision 13. ML14167A316. Washington, DC: U.S. Nuclear Regulatory Commission. 2014.
_____. NUREG–0383, “Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance.” Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.
_____. NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology.” Oak Ridge, Tennessee: Science Applications International Corporation. U.S. Nuclear Regulatory Commission. 2001.
_____. NUREG/CR-5661, “Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages.” ORNL/TM-11936. Oak Ridge, TN: Oak Ridge National Laboratory. U.S. Nuclear Regulatory Commission. 1997.</td></tr>
</table>

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.1 Drawings		Content of drawings	Information available on many NRC-certified package designs, particularly the TN-FSV and NAC-LWT packages for transport of TRISO fuel (NRC, 2021, 2014, 2013); Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for examining the content of engineering drawings, as well as the description of materials in package designs.
7.4.2 Codes and Standards	Usage and endorsement	Codes and standards used for the package design and construction	Codes and standards are available (NRC, 2021, 2014, 2013); however, it is uncertain whether those standards would apply to new materials potentially to be used for package design and fabrication for transport of TRISO fuel	Applicability of codes and standards for package design and fabrication with new materials is to be evaluated	None identified. The review method calls for verification of the codes and standards for packaging components important to safety. Codes and standards to be used are expected to be defined, or the technical basis be provided for the adequacy of alternative codes and standards.
	ASME code components	Construction of ASME code components	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Code case use/acceptability	Acceptability of ASME code cases	Specific code case referenced is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Non-ASME code components	Construction of non-ASME code components	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.3 Weld Design and Inspection	Weld Design and Inspection	Welding criteria and weld procedure qualification requirements	Detailed design and construction information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that standard welding processes are adequate for package design and fabrication for transport of TRISO fuel. If new technologies were used in the design and fabrication of welds, the SRP calls for examination of compliance with any established codes and standards proposed in the application on design and construction.

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Moderator exclusion for commercial spent nuclear fuel packages under hypothetical accident conditions	Not applicable to packages for spent TRISO fuel transportation	Not applicable	Not applicable	Not applicable
7.4.4 Mechanical Properties	Tensile properties	Acceptability of material tensile properties	Mechanical properties for commonly used packaging materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. It is assumed that commonly used packaging materials may be also adequate for package design and fabrication for transport of TRISO fuel. If alternative or new materials were required in the design and fabrication of transportation packages, the SRP calls for examination of the adequacy of information in the application related to mechanical properties of those alternative materials.

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Fracture resistance	Acceptability of material fracture toughness	Mechanical properties for commonly used packaging materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Tensile properties and creep of aluminum alloys at elevated temperatures	Acceptability of the tensile properties and creep of aluminum alloys	Mechanical properties for commonly used aluminum alloys are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Impact limiters	Acceptability of the mechanical properties of the impact limiter materials	Mechanical properties for commonly used impact limiter materials are available	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.5 Thermal Properties of Materials		Thermal properties of package materials; Effect of degradation and anisotropic dependencies of thermal properties	Thermal properties for commonly used packaging materials are available; Detailed evaluation of package components and fuels is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.4.6 Radiation Shielding	Neutron-shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Gamma-shielding materials	Compositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performance	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.7 Criticality Control	Neutron-absorbing (poison) material specification	Chemical composition, physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; Qualification testing	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Computation of percent credit for boron-based neutron absorbers	Level of credit allowed for absorber materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Qualifying properties not associated with attenuation	Qualification of absorber material properties not associated with neutron attenuation	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.8 Corrosion Resistance	Environments	Range of environmental conditions encountered for package components	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Carbon and low-alloy steels	Environment dependencies of corrosion rate; Coatings for corrosion prevention	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
	Austenitic stainless steel	Localized corrosion and chloride-induced stress corrosion cracking in chloride-containing environments; Intergranular corrosion and stress corrosion cracking in sensitized stainless steel	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.9 Protective Coatings	Review guidance	Coating specifications	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Scope of coating application	Purpose of the coating, lists the components to be coated, and the expected environmental conditions	Detailed design information is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	Coating selection	Coating manufacturer, type of primers and topcoat, coating thickness, and ability of the coating to withstand the in-service conditions	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Coating qualification testing	Qualification testing for coating performance in accordance with several standard ASTM (and possibly other) tests	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. The SRP calls for evaluating any qualification testing for the demonstration of coating performance.
7.4.10 Content Reactions	Flammable and explosive reactions	Effects of flammable and explosive reactions among the content materials	Information available on NRC-certified TN-FSV and NAC-LWT packages (NRC, 2021, 2014); Detailed evaluation of package components and fuels is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Content chemical reactions,	Effects of chemical reactions, outgassing, and corrosion among	TRISO fuel discharged from fluoride salt-cooled high-temperature reactors	Material performance of FHR fuel with residual salt	Additional guidance may need to be developed to address corrosion of non-fuel hardware

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
	outgassing, and corrosion	the contents and between the contents and the package components	(FHR) may contain residual salt coolant. Radiolysis of solid fluoride salts in radiation fields will generate fluorine gas that is toxic and potentially corrosive (Forsberg and Peterson, 2015).	material is to be evaluated	associated with TRISO fuel. The SRP calls for examining that corrosion wastage will not lead to a loss of intended functions; however, for non-fuel hardware the current review method is limited to guidance for the examination of corrosion of hardware components associated with stainless steel or zirconium alloy-clad UO_2 fuels.

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel				
Areas of review (NUREG-2216, Chapter 7)	**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.11 Radiation Effects	Effects of radiation on the performance of the package materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies. Commonly used packaging materials may be also adequate for package design and fabrication for transport of TRISO fuel. If alternative or new materials were required in the design and fabrication of transportation packages, the SRP calls for examination of the adequacy of information in the application related to radiation effects on those alternative materials.
7.4.12 Package Contents	Chemical and physical form of the package contents; Effects of corrosion, chemical reactions, and radiation on the properties of the contents	Detailed evaluation of package components and fuels is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.13 Fresh (Unirradiated) Fuel Cladding	Not applicable to spent TRISO fuel	Not applicable	Not applicable	Not applicable

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel

Areas of review (NUREG-2216, Chapter 7)		Key information to be reviewed	Information availability	Potential information needs	Potential guidance gaps
7.4.14 Spent Nuclear Fuel	Spent fuel classification	Classification of damaged, undamaged, and intact fuel	Information is expected to be available at the time of an application	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Uncannistered spent fuel	Cladding alloys and maximum fuel burnup; Cladding mechanical properties; Effective cladding thickness; Maximum cladding temperature; Thermal cycling during loading operations; Composition of the cover gas; High burnup fuel monitoring and assessment; Release fractions	Although degradation of TRISO fuel has not been reported associated with transportation conditions, information in the literature is limited	Performance of the coating layers on TRISO fuel under transportation environments is to be evaluated	Additional guidance may need to be developed to address the mechanical properties of coating layers of TRISO fuel. The current review method is limited to guidance for the performance of zirconium alloy, aluminum alloy, or stainless steel-clad UO_2 fuels, which is not applicable to TRISO fuel. It is necessary to examine whether similar or equivalent cladding functions are required in TRISO fuel.
	Cannistered spent fuel	Performance of the fuel can for damaged fuel	Detailed evaluation of the fuel can performance is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent TRISO fuel					
Areas of review (NUREG-2216, Chapter 7)		**Key information to be reviewed**	**Information availability**	**Potential information needs**	**Potential guidance gaps**
7.4.15 Bolting Material		Material properties of the bolting; Effects of corrosion, chemical reactions, and radiation on the bolting materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
7.4.16 Seals	Metallic seals	Material properties of metallic seals; Effects of corrosion, chemical reactions, and radiation on the seal materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies
	Elastomeric seals	Material properties of elastomeric seals; Effects of corrosion, chemical reactions, and radiation on the seal materials	Detailed evaluation of packaging materials is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies

Forsberg, C. and P.F. Peterson. "Spent Nuclear Fuel and Graphite Management for Salt-Cooled Reactors: Storage, Safeguards, and Repository Disposal." Nuclear Technology. Vol. 191. pp. 113–121. 2015.

NRC. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9225 for the NAC-LWT Package." Revision 71. ML21078A200. Washington, DC: U.S. Nuclear Regulatory Commission. 2021.

_____. "Certificate of Compliance for Radioactive Materials Packages, Certificate No. 9253 for the TN-FSV Package." Revision 13. ML14167A316. Washington, DC: U.S. Nuclear Regulatory Commission. 2014.

_____. NUREG–0383, "Directory of Certificates of Compliance for Radioactive Materials Packages, Certificates of Compliance." Volume 2, Revision 28. ML13309A031. Washington, DC: U.S. Nuclear Regulatory Commission. 2013.