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

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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 hightemperature 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 UO₂ 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 UO₂ 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.

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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

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QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

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1 INTRODUCTION

1.1 <u>Background</u>

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 ²³⁵U¹. 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 ²³⁵U 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

¹Nuclear fuel enriched to less than 20 weight percent ²³⁵U 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 nonfuel hardware components¹ 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 hightemperature 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 <u>Mechanical Properties of Cladding or Coatings</u>

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

¹Non-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.

Advanced² stainless steel cladding materials (i.e., D9³ and HT9⁴) were developed for sodiumcooled 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-coated⁵ 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 UO₂ 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 UO₂ 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 ²³⁵U. 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 UO₂ fuel, but the compositions of ARFs are different. For example, TRISO fuel may be uranium oxycarbide

²The 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.

³D9 is a titanium modified variant of Type 316 stainless steel (Makenas, 1986). ⁴HT9 is a 12 percent chromium, 1 percent molybdenum ferritic/martensitic stainless steel (Brown et al., 2012).

⁵In addition to the SiC coating 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 UO₂. 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.

⁶Uranium oxycarbide (UCO) is a short-hand notation for a mixture of uranium dioxide (UO₂) 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 (B₄C). This type of control element is consistent with the regulatory guidance¹. 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 (Eu₂O₃) and gadolinium oxide (Gd₂O₃) control rods in comparison to B₄C rods. Guo et., al (2020) also describe the use of burnable poisons consisting of neptunium, depleted B₄C, 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

¹Section 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 B₄C (Qualls et al., 2016). Gougar (2014) describes metal-clad control rods made of Incoloy-800H (nickel-iron-chromium alloy) tubes enclosing B₄C annular graphite compacts as well as absorbers made of pyrolytic carbon-coated spheres of B₄C 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 technologyneutral 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 <u>Mechanical Properties of Cladding or Coatings</u>

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 uncanned² 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 advanced³ 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, hightemperature 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.

³Advanced 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 UO₂ 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 UO₂, PuO₂, 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 TRISOcoated 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 mmdiameter [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 ²³⁵U assay (or concentration) above 5 percent but below 20 percent. Whereas existing LWRs operate on low enriched uranium (LEU) with up to 5 percent ²³⁵U, 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 UO₂ fuel enriched up to 5.0 weight percent ²³⁵U 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 UO₂ 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 UO₂ 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 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 UO₂ low enriched uranium (LEU) fuel with up to 5 percent ²³⁵U 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 UO₂ fuel enriched up to 5.0 weight percent ²³⁵U; 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 UO₂ 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.

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APPENDIX I

TRANSPORTATION OF FRESH FUEL

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages								
for	for fresh metal fuel							
Areas of	f review	Key information	Information	Potential	Potential guidance			
(NUREG-221	6, Chapter 2)	to be reviewed	availability	information needs	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. Info for	Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel						
Areas of	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 2)	to be reviewed	availability	Information needs	gaps		
	Tamper-	The package	Detailed package	None expected	None identified;		
	indicating	closure system	design information is		General acceptance		
	feature	must incorporate a	expected to be provided		criteria not impacted		
		tamper-indicating	by the applicant		by specifics of		
		feature			technologies		
	Positive closure	The package	Detailed package	None expected	None identified;		
		closure system	design information is		General acceptance		
		must include a	expected to be provided		criteria not impacted		
		positive fastening	by the applicant		by specifics of		
		device	, , , , , , , , , , , , , , , , , , , ,		technologies		
	Package valve	A package valve	Detailed package	None expected	None identified;		
	Ŭ	or other device	design information is	•	General acceptance		
		must be protected	expected to be provided		criteria not impacted		
		against	by the applicant		by specifics of		
		unauthorized			technologies		
		operation					
2.4.3 Lifting and	Lifting devices	Lifting devices	Detailed design	None expected	None identified:		
Tie-Down	5	must be designed	information is expected	I	General acceptance		
Standards for All		in accordance with	to be provided by the		criteria not impacted		
Packages		10 CFR 71 45(a)	applicant		by specifics of		
1 donagee			approant		technologies		
	Tie-down	Tie-down devices	Detailed design	None expected	None identified		
	devices	must be designed	information is expected		General accentance		
	001003	in accordance with	to be provided by the		criteria not impacted		
		10 CER 71 45(b)	applicant		by specifics of		
		10 CFR / 1.45(D)	applicant				
					lechnologies		

Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages					
for	fresh metal fuel	-			
Areas o	f review	Key information	Information	Potential	Potential guidance
(NUREG-221	6, Chapter 2)	to be reviewed	availability	information needs	gaps
2.4.4 General	Evaluation by	Elements of the	Specific analyses used	None expected	None identified;
Considerations	analysis	analysis used for	to evaluate the package		General acceptance
for Structural		structural	are expected to be		criteria not impacted
Evaluation of		evaluation	provided by the		by specifics of
Раскаділд	Freebootton has		applicant	No	technologies
	Evaluation by	Elements of the	Specific tests used to	None expected	None Identified;
	test	test used for	evaluate the package		General acceptance
		structurar	are expected to be		by aposition of
		evaluation	applicant		technologies
2.4.5 Normal	Heat	Maximum	Structural performance	None expected	None identified:
Conditions of	Tieat	temperature	of the package under	None expected	General accentance
Transport		maximum	the heat-loading		criteria not impacted
ridhoport		pressure, and	condition is expected to		by specifics of
		thermal stress	be provided by the		technologies
		under the heat-	applicant		0
		loading condition			
	Cold	Maximum	Structural performance	None expected	None identified;
		temperature,	of the package under		General acceptance
		minimum internal	the cold condition is		criteria not impacted
		pressure, and	expected to be provided		by specifics of
		residual stress	by the applicant		technologies
		under the cold			
		condition			
	Reduced	Effects of reduced		None expected	None Identified;
	external	external pressure	external pressure is		General acceptance
	pressure		described and		by specifies of
		of the nackade	evaluated by the		technologies
		and the	applicant		teennologies
		containment	approxim		
		system			

Table I-1. Inf for	Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	6, Chapter 2)	to be reviewed	availability	information needs	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. Info for	Table I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel					
Areas of	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	6, Chapter 2)	to be reviewed	availability	information needs	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. Info for	ble I-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh metal fuel						
Areas o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 2)	to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance		
(NUREG-2216, Chapter 2)	to be reviewed	availability	information needs	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 That 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	Key information	Information	Potential	Potential guidance	
(NUREG-2216, Chapter 2)	to be reviewed	availability	information needs	gaps	
	applicability of 10 CFR 71.61)				
2.4.9 Air Transport of Plutonium	Not applicable	Not applicable	Not applicable	Not applicable	
	because air				
	transport of				
	plutonium is not				
	anticipated				
NRC. NUREG–0383, "Directory of Certificat	es of Compliance for Radi	oactive Materials Packages, Cert	ificates of Compliance." Vo	lume 2, Revision 28.	
ML13309A031. Washington, DC: U.S. Nuc	ear Regulatory Commissio	on. 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. "App	ication for Certificate of Co	ompliance for the Traveller PWR	Fuel Shipping Package." S	afety Analysis Report,	
Revision 1. ML19308C710. Westinghouse Electric Company LLC. 2019.					

Table I-2. Info	Table I-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation					
pac	kages for fresh n	netal fuel				
Areas of	[:] review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	6, Chapter 3)	to be reviewed	availability	information needs	gaps	
3.4.1 Description	Packaging	Drawings and	Information available on	None expected	None identified;	
of the Thermal	design features	description of the	many NRC-certified		General acceptance	
Design		thermal features	package designs		criteria not impacted	
			(NRC, 2013); detailed		by specifics of	
			description of thermal		technologies	
			features is expected to			
			be provided by the			
			applicant			
	Codes and	Codes and	Codes and standards	None expected	None identified;	
	standards	standards used for	used to design the			
		the thermal design	to be evailable		criteria not impacted	
		the peekege	to be available			
	Contant haat		Detailed design	None expected	Nono identified:	
	Load	heat load:	information is expected	None expected	Ceneral accentance	
	specification	Methods and	to be provided by the		criteria not impacted	
	specification	codes used to	applicant		by specifics of	
		determine content	applicant		technologies	
		decay heat loads				
	Summary	Maximum.	Detailed design	None expected	None identified:	
	tables of	minimum, and	information is expected		General acceptance	
	temperatures	allowable	to be provided by the		criteria not impacted	
		temperatures of	applicant		by specifics of	
		package			technologies	
		components for			-	
		normal and				
		accident				
		conditions				

Table I-2. Information to be reviewed and potential gaps for evaluating thermal performance of transportation					
pac	kages for fresh n	netal fuel			
Areas of	review	Key information	Information	Potential	Potential guidance
(NUREG-2216	5, Chapter 3)	to be reviewed	availability	information needs	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. Info	formation to be reviewed and potential gaps for evaluating thermal performance of transportation					
packages for fresh metal fuel						
Areas of review		Key information	Information	Potential	Potential guidance	
(NUREG-2216, Chapter 3)		to be reviewed	availability	information needs	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						
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pac	kages for fresh n	Key information	Information	Potential	Potential guidance	
(NUREG-2216	6. Chapter 3)	to be reviewed	availability	information needs	gans	
	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	Areas of review Key information Information Potential Potential guidan						
(NUREG-2216	6, Chapter 3)	to be reviewed	availability	information needs	gaps		
	operating	pressure under the	to be provided by the		by specifics of		
	pressure	heat condition	applicant		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 rev	iewed and potential	gaps for evaluating them	mal performance of t	ransportation			
packages for fresh m	ietal fuel			-			
Areas of review	Key information	Information	Potential	Potential guidance			
(NUREG-2216, Chapter 3)	to be reviewed	availability	information needs	gaps			
Janney, D.E. and S.L. Hayes. "Experimental	y Known Properties of U-1	10Zr alloys: A critical Review." N	uclear Technology. Vol. 20	3. pp.109–128. 2018.			
Janney, D.E., S.L. Hayes, and C.A. Adkins. "	A Critical Review of the Ex	xperimentally Known Properties of	of U-Pu-Zr Alloys. Part 1: P	hases and Phase			
Diagrams." Nuclear Technology. Vol. 205. p	p.1,387–1,415. 2019.						
Janney, D.E., S.L. Hayes, and C.A. Adkins. "	A Critical Review of the Ex	xperimentally Known Properties of	of U-Pu-Zr Alloys. Part 2: 1	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, Includir	ng Those with Minor Actinid	es (Np, Am, Cm), Rare-			
earth Elements (La, Ce, Pr, Nd, Gd), and Y."	INL/EXT-15-36520 Revisi	ion 3 Part 1. Idaho Falls, Idaho:	Idaho National Laboratory.	2018a.			
Janney, D.E. "Metallic Fuels Handbook, Part	2: Elements and Alloys no	ot Based on U-Zr, Pu-Zr, U-Pu, o	r U-Pu-Zr." INL/EXT-15-365	520 Revision 3 Part 2. Idaho			
Falls, Idaho: Idaho National Laboratory. 2018	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 Certificate	s of Compliance for Radio	oactive Materials Packages, Certi	ficates of Compliance." Vo	lume 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	Key information	Information	Potential	Potential guidance	
(NUREG-2216	6, Chapter 4)	to be reviewed	availability	information needs	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	None expected	None identified; General acceptance criteria not impacted by specifics of technologies	
	Codes and standards Special requirements for damaged spent nuclear fuel	Codes and standards used for the containment design of the package Not applicable to fresh metal fuel	Codes and standards used to design the package are expected to be available Not applicable	None expected Not applicable	None identified; General acceptance criteria not impacted by specifics of technologies Not applicable	

Table I-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation								
Areas of	Areas of review Key information Information Potential Distance							
(NUREG-2216	6, Chapter 4)	to be reviewed	availability	information needs	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						
рас	kages for fresh	metal fuel		1		
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	5, Chapter 4)	to be reviewed	availability	information needs	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	Type B transportation	Releasable source term, maximum	Information available on many NRC-certified	Release rate calculations and	None identified. The review method calls for	
Evaluation under	packages	permissible	package designs	criteria used to	ensuring the applicant	
Normal		release rate,	(NRC, 2013;	verify cladding	calculated the	
		maximum	Westinghouse Electric	welds and fuel pins	maximum permissible	
Transport		permissible	Company LLC., 2019)	integrity to ensure	release rate and	
		leakage rate, and	and previous		haximum permissible	
		conversion to the	testing of metal fuel size	transportation		
			(Burkes et al. 2002):		NIA 5 The evicting	
		leakage rate	(Burkes et al., 2009);	package of the	roviow mothed is	
		calculated for	Detailed containment	soulum-containing	review method is	

Table I-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation							
pac	packages for fresh metal fuel						
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	6, Chapter 4)	to be reviewed	availability	information needs	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 packages for fresh metal fuel						
Areas of (NUREG-2216	review 6, 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	Key information	Information	Potential	Potential guidance
(NUREG-2216, Chapter 4)	to be reviewed	availability	information needs	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. In	Table I-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation						
ра	ckages for fresh	metal fuel	1	1			
Areas o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	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. Int	Table I-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation						
ра	ckages for fresh	metal fuel					
Areas o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	gaps		
	Gamma	Gamma source	Metal fuel fabricated	Source-term	None identified. The		
	sources	strengths and	with reprocessed	specification	review method calls for		
		spectra for the	uranium could possibly	including gamma	ensuring the		
		package contents	contain radioactive sources; Although ASTM C996 (ASTM, 2021) is frequently applied, information in the literature is limited	sources and their energies arising from fission products from reprocessed uranium or other impurities is to be evaluated	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						
ра	ckages for fresh	metal fuel		1	1	
Areas o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	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 r packages for freek	I-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation						
Areas of review	Key information	Information	Potential	Potential guidance			
(NUREG-2216, Chapter 5)	to be reviewed	availability	information needs	gaps			
Code input and	Key input data	Detailed shielding	None expected	None identified;			
output data	and output files for	evaluation is expected		General acceptance			
	the shielding	to be provided by the		criteria not impacted by			
	evaluations	applicant		specifics of			
				technologies			
Fluence-rate-	Accuracy and	Detailed shielding	None expected	None identified;			
to-radiation-	acceptance of the	evaluation is expected		General acceptance			
level	conversion factors	to be provided by the		criteria not impacted by			
conversion		applicant		specifics of			
factors				technologies			
External	External radiation	Detailed shielding	None expected	None identified;			
radiation levels	levels under	evaluation is expected		General acceptance			
	normal and	to be provided by the		criteria not impacted by			
	accident	applicant		specifics of			
	conditions			technologies			
Confirmatory	Rigor of the	Detailed shielding	None expected	None identified;			
analyses	confirmatory	evaluation is expected		General acceptance			
	analyses	to be provided by the		criteria not impacted by			
		applicant		specifics of			
				technologies			

ASTM. C996. "Standard Specification for Uranium Hexafluoride Enriched to Less Than 5% ²³⁵U" <<u>https://compass.astm.org/EDIT/html_annot.cgi?C996+20</u>> (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. <<u>https://rampac.energy.gov/docs/default-source/certificates/1019342.pdf>.(Accessed</u> 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	Key information	Information	Potential	Potential guidance	
(NUREG-2216	6, Chapter 6)	to be reviewed	availability	information needs	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 <i>k</i> 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 mater	ial 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. Info	ble I-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh metal fuel					
Areas of (NUREG-221	f review 6, 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 <i>k</i> 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	Key information	Information	Potential	Potential guidance		
(NUREG-2216	6, Chapter 6)	to be reviewed	availability	information needs	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						
Key information	Information	Potential	Potential guidance			
to be reviewed	availability	information needs	gaps			
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			
	viewed and potentia netal fuel Key information to be reviewed Benchmarking computer codes for criticality calculations against fitting critical experiments	viewed and potential gaps for evaluating critKey information to be reviewedInformation availabilityBenchmarking computer codes for criticality calculations against fitting critical experimentsInformation 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).	viewed and potential gaps for evaluating criticality performance of netal fuelKey information to be reviewedInformation availabilityPotential information needsBenchmarking computer codes for criticality calculations against fitting critical experimentsInformation 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 performance of Potential information needsViewed and potential 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 performation needs criticality codes and methods and 20 weight percent (Jarrell, 2018).			

Table I-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh metal fuel						
Areas of review		Key information	Information	Potential	Potential guidance	
(NUREG-2216	6, Chapter 6)	to be reviewed	availability	information needs	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								
р	packages for fresh metal fuel							
Areas	of review	Key information	Information	Potential	Potential guidance			
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	gaps			
7.4.1 Drawings		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 Packages for fresh metal fuel							
Areas	of review	Key information	Information	Potential	Potential guidance		
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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		Key information	Information	Potential	Potential guidance	
(NUREG-2216, Chapter 7)		to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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					
р	ackages for fresh	metal fuel			•
Areas	of review	Key information	Information	Potential	Potential guidance
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance	
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	gaps	
7.4.5 Thermal P Materials	roperties of	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	
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							
р	packages for fresh metal fuel						
Areas of review		Key information	Information	Potential	Potential guidance		
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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					
р	ackages for fresh	metal fuel	· · ·		
Areas of review		Key information	Information	Potential	Potential guidance
(NUREG-22	16, Chapter 7)	to be reviewed	availability	Information needs	gaps
	associated with		expected to be provided		by specifics of
	allenuation	attonuction	by the applicant		lechnologies
7 / 9	Environmonto	Pango of	Dotailed design	None expected	None identified:
Corrosion		environmental	information is expected	None expected	General accentance
Resistance		conditions	to be provided by the		criteria not impacted
resistance		encountered for	applicant		by specifics of
		package	apprioant		technologies
		components			loomoogioo
	Carbon and low-	Environment	Detailed evaluation of	None expected	None identified;
	alloy steels	dependencies of	packaging materials is		General acceptance
		corrosion rate;	expected to be provided		criteria not impacted
		Coatings for	by the applicant		by specifics of
		corrosion			technologies
		prevention			
	Austenitic	Localized	Detailed evaluation of	None expected	None identified;
	stainless steel	corrosion and	packaging materials is		General acceptance
		chloride-induced	expected to be provided		criteria not impacted
		stress corrosion	by the applicant		by specifics of
					technologies
		environmente:			
		Intergranular			
		corrosion and			
		stress corrosion			
		cracking in			
		sensitized			
		stainless steel			
7.4.9	Review	Coating	Detailed design	None expected	None identified;
Protective	guidance	specifications	information is expected		General acceptance
Coatings			to be provided by the		criteria not impacted
			applicant		

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel packages for fresh metal fuel					
Areas of review		Key information to be reviewed	Information availability	Potential information needs	Potential guidance
			,		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					
Areas	of review	Key information	Information	Potential	Potential guidance
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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					
	ackages for fresh	Kov information	Information	Potential	Potontial guidanco
(NUREG-22	16 Chanter 7)	to be reviewed	availability	information needs	rotential guidance
					of corrosion of hardware components associated with stainless steel or zirconium alloy-clad UO ₂ fuels.
7.4.11 Radiation	n 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 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						
р	ackages for fresh	metal fuel				
Areas	of review	Key information	Information	Potential	Potential guidance	
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	gaps	
7.4.13 Fresh (U	nirradiated) Fuel	Mechanical	Some data are available	Advanced cladding	Additional guidance	
Cladding		properties of the	on metal fuel cladding	material properties	may need to be	
		cladding materials;	materials (Carmack et	that can be used to	developed to	
		Sources of	al., 2009; Garner, 1993;	achieve high burnup	address the	
		cladding materials	IAEA, 2012)	are to be evaluated,	mechanical	
		data		especially material	properties of	
				performance data	advanced cladding	
				under the influence of	materials for metal	
				irradiation	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	Spent fuel	Not applicable to	Not applicable	Not applicable	Not applicable	
Nuclear Fuel	classification	fresh metal fuel				
	Uncanned spent	Not applicable to	Not applicable	Not applicable	Not applicable	
	fuel	fresh metal fuel				
	Canned spent	Not applicable to	Not applicable	Not applicable	Not applicable	
	fuel	fresh metal fuel				
1						

Table I-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh metal fuel packages for fresh metal fuel						
Areas	of review	Key information	Information	Potential	Potential guidance	
(NUREG-2216, Chapter 7) 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	gapsNone identified;General acceptancecriteria not impactedby specifics oftechnologies	
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 Key information Information Potential Potential guidance (NUREG-2216, Chapter 7) to be reviewed availabilitv information needs 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), Rareearth 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 For fresh TRISO fuel						
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	6, Chapter 2)	to be reviewed	availability	information needs	gaps	
2.4.1 Description	General	Drawings and	Information available on	None expected	None identified;	
of Structural		descriptive	many NRC-certified		General acceptance	
Design		information	package designs,		criteria not impacted	
		including weights	particularly the Versa-		by specifics of	
		and centers of	Pac package for		technologies	
		gravity	transport of fresh			
			TRISO fuel (NRC, 2013;			
			Century Industries,			
			2009, 2010); Detailed			
			description of structural			
			design is expected to be			
			provided by the			
			applicant			
	Identification of	Codes and	Codes and standards	None expected	None identified;	
	codes and	standards used for	used to design the		General acceptance	
	standards for	the package	package are expected		criteria not impacted	
	package	design and	to be available		by specifics of	
	design	fabrication			technologies	
2.4.2 General	Minimum	The smallest	Specific package	None expected	None identified;	
Requirements for	package size	overall dimension	dimensions are		General acceptance	
All Packages		of the package	expected to be provided		criteria not impacted	
		must not be less	by the applicant		by specifics of	
		than 10 cm [4 in]			technologies	
	Tamper-	The package	Detailed package	None expected	None identified;	
	indicating	closure system	design information is		General acceptance	
	feature	must incorporate a	expected to be provided		criteria not impacted	
		tamper-indicating	by the applicant		by specifics of	
		feature			technologies	

Table I-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel For fresh TRISO fuel						
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	5, Chapter 2)	to be reviewed	availability	information needs	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						
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for fresh TRISO fuel						
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	<u>, Chapter 2)</u>	to be reviewed	availability	information needs	gaps	
2.4.5 Normal	Heat	Maximum	Structural performance	None expected	None identified;	
Conditions of		temperature,	of the package under		General acceptance	
Transport		maximum	the heat-loading		criteria not impacted	
		pressure, and	condition is expected to		by specifics of	
		thermal stress	be provided by the		technologies	
		under the heat-	applicant			
		loading condition				
	Cold	Maximum	Structural performance	None expected	None identified;	
		temperature,	of the package under		General acceptance	
		minimum	the cold condition is		criteria not impacted	
		internal pressure,	expected to be provided		by specifics of	
		and residual stress	by the applicant		technologies	
		under the				
		cold condition				
	Reduced	Effects of reduced	Effects of reduced	None expected	None identified;	
	external	external pressure	external pressure is		General acceptance	
	pressure	on the internal and	expected to be		criteria not impacted	
		external	described and		by specifics of	
		pressures of the	evaluated by the		technologies	
		package and the	applicant			
		containment				
		system				
	Increased	Effects of	Effects of increased	None expected	None identified;	
	external	increased	external pressure is		General acceptance	
	pressure	external pressure	expected to be		criteria not impacted	
		on the internal and	described and		by specifics of	
		external	evaluated by the		technologies	
		pressures of the	applicant			
		package and the				
		containment				
		system				

Table I-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel Information packages						
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	5, Chapter 2)	to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance
(NUREG-2216	5, Chapter 2)	to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance	
(NUREG-2216, Chapter 2)	to be reviewed	availability	information needs	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							
pac	packages for fresh TRISO fuel						
	review	to be reviewed	Information	Potential	Potential guidance		
2 4 1 Description	Dookoging	Drowings and	availability	None expected	yaps None identified:		
of the Thermal	design features	description of the		None expected	General acceptance		
	uesign leatures	thermal features	certified nackade		criteria not impacted		
Design			designs particularly		by specifics of		
			the Versa-Pac		technologies		
			package for transport		g		
			of fresh TRISO fuel				
			(NRC, 2013; Century				
			Industries, 2009,				
			2010); Detailed				
			description of thermal				
			teatures is expected				
			to be provided by the				
	Codes and	Codes and		None expected	None identified:		
	standards	standards used for	used to design the	None expected	General accentance		
	Standards	the thermal design	package are expected		criteria not impacted		
		and evaluation of	to be available		by specifics of		
		the package			technologies		
	Content heat	Maximum decay	Detailed design	None expected	None identified;		
	load	heat load;	information is		General acceptance		
	specification	Methods and	expected to be		criteria not impacted		
		codes used to	provided by the		by specifics of		
		determine content	applicant		technologies		
	Summony	Movimum	Detailed design	None expected	Nono identified:		
	tables of	minimum and	information is	None expected	General accentance		
	temperatures	allowable	expected to be		criteria not impacted		
		temperatures of	provided by the		by specifics of		
		package	applicant		technologies		
		components for			Ū		
		normal and					

Table I-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation						
Areas of review		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							
pac	packages for fresh TRISO fuel						
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	5, Chapter 3)	to be reviewed	availability	information needs	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 Surface Temperatu	Accessible ures	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						
pac	kages for fresh T	RISO fuel			1	
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	6, Chapter 3)	to be reviewed	availability	information needs	gaps	
		calculating the	to be provided by the		by specifics of	
		accessible	applicant		technologies	
		surface				
		temperatures				
3.4.5 Thermal	Heat and cold	Maximum	Detailed thermal	None expected	None identified;	
Evaluation Under		accessible	evaluation is expected		General acceptance	
Normal		surface	to be provided by the		criteria not impacted	
Conditions of		temperatures;	applicant		by specifics of	
Transport		Maximum			technologies	
		temperatures of				
		package				
		components under				
		the heat condition;				
		Minimum				
		temperatures of				
		package				
		components under				
		the cold condition				
	Maximum	The maximum	Detailed thermal	None expected	None identified;	
	normal	normal operating	evaluation is expected		General acceptance	
	operating	pressure under the	to be provided by the		criteria not impacted	
	pressure	heat condition	applicant		by specifics of	
					technologies	
3.4.6 Thermal	Initial	Initial conditions of	Detailed thermal	None expected	None identified;	
Evaluation Under	conditions	the package	evaluation is expected		General acceptance	
Hypothetical			to be provided by the		criteria not impacted	
Accident			applicant		by specifics of	
Conditions					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 pressure	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.						

Table I-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh TRISO fuel						
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	, Chapter 4)	to be reviewed	availability	information needs	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 Special requirements for damaged	Codes and standards used for the containment design of the package Not applicable to fresh TRISO fuel	Codes and standards used to design the package are expected to be available Not applicable	None expected Not applicable	None identified; General acceptance criteria not impacted by specifics of technologies Not applicable	
	fuel					

Table I-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation packages for fresh TRISO fuel						
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	, Chapter 4)	to be reviewed	availability	information needs	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						
pac	packages for fresh TRISO fuel					
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	6, Chapter 4)	to be reviewed	availability	information needs	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						
pac	packages for fresh TRISO fuel					
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	6, Chapter 4)	to be reviewed	availability	information needs	gaps	
Hypothetical		maximum	under hypothetical		by specifics of	
Accident		permissible	accident conditions;		technologies	
Conditions		leakage rate, and	Detailed package		_	
		conversion to the	containment evaluation			
		reference air	under hypothetical			
		leakage rate	accident conditions is			
		calculated for	expected to be provided			
		hypothetical	by the applicant			
		accident				
		conditions in				
		accordance with				
		ANSI N14.5				
	Spent nuclear	Not applicable to	Not applicable	Not applicable	Not applicable	
	fuel	fresh TRISO fuel				
	transportation					
	packages					
	Compliance	Packages must be	Detailed design	None expected	None identified;	
	with	designed to satisfy	information is expected		General acceptance	
	containment	the containment	to be provided by the		criteria not impacted	
	design criteria	requirements of 10	applicant		by specifics of	
	-	CFR 71.51(a)(2)			technologies	
		under hypothetical			-	
		accident				
		conditions				
ANSI. ANSI N14.5-20	14, "American Nation	al Standard for Radioactiv	e Materials–Leakage Tests on Pa	ackages for Shipment." New	York, New York: American	
National Standards In	stitute. 2014.	t for the Conturn Induction	Varia Das Chinning Cantainan"	Devision 0 MI 002224070	Printel Vincinia, Contum	
Industries 2009	salety Analysis Repor	t for the Century Industries	versa-Pac Snipping Container.	Revision 0. ML092321070.	Bristol, Virginia: Century	
Century Industries. 2009. Century Industries Versa-Pac Shipping Container." Revision 3. ML101110135. Bristol. Virginia: Century						
Industries. 2010.	, , , , , , , , , , , , , , , , , , , ,	- ,			, , , , , , , ,	
Eidelpes, E., J.J. Jarre	ell, H.E. Adkins, B.M.	Hom, J.M. Scaglione, R.A.	Hall, and B.D. Brickner. "UO2 H	ALEU Transportation Packag	ge Evaluation and	
Recommendations."	NL/EXI-19-56333. Id	tano Falls, Idaho: Idaho N tos of Compliance for Bod	lational Laboratory. 2019.	rtificator of Compliance " Mal	umo 2 Povision 29	
ML13309A031. Wash	ington, DC: U.S. Nuc	clear Regulatory Commissi	ion. 2013.	runcates of Compliance. Vol	unie 2, Nevision 20.	

Table I-10. Inf	Table I-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh TRISO fuel						
Areas o	of review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	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. Inf	Table I-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation packages for fresh TRISO fuel						
Areas o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	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								
ра	ckages for fresh 1	TRISO fuel						
Areas o	of review	Key information	Information	Potential	Potential guidance			
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	gaps			
		accident			by specifics of			
		conditions			technologies			
	Confirmatory	Rigor of the	Detailed shielding	None expected	None identified;			
	analyses	confirmatory	evaluation is expected		General acceptance			
		analyses	to be provided by the		criteria not impacted			
		, ,	applicant		by specifics of			
					technologies			
ASTM. C996. "Standa	ard Specification for Ura	anium Hexafluoride Enrich	ed to Less Than 5% ²³⁵ U" < <u>https:</u>	//compass.astm.org/EDIT/h	tml_annot.cgi?C996+20>			
(Accessed May 27, 2	021). 2021.							
Century Industries. "	Century Industries. "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container." Revision 3. ML101110135. Bristol, Virginia: Century							
Industries. 2010.	Industries. 2010.							
NRC. NUREG-0383	, "Directory of Certificat	tes of Compliance for Radi	oactive Materials Packages, Cert	ificates of Compliance." Vo	lume 2, Revision 28.			
ML13309A031. Was	hington, DC: U.S. Nuc	lear Regulatory Commissio	on. 2013.					

Table I-11. Info page	Table I-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel						
Areas of	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 6)	to be reviewed	availability	information needs	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	Information availability	Potential information needs	Potential guidance		
(,				by specifics of technologies		
6.4.2 Fissile mater	ial 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	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 6)	to be reviewed	availability	information needs	gaps		
	Demonstration of maximum reactivity	Analyses demonstrate the maximum <i>k</i> 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							
Areas of review		Key information	Information availability	Potential information needs	Potential guidance		
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	Key information	Information	Potential	Potential guidance		
(NUREG-2216	5, Chapter 6)	to be reviewed	availability	information needs	gaps		
(NUREG-2210 6.4.6 Benchmark Evaluations	5, Chapter 6) Experiments and applicability	to be reviewed Benchmarking computer codes for criticality calculations against fitting critical experiments	availability 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)	information needs 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	gaps 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		Key Information	Information	Potential	Potential guidance		
(NUREG-2216	6, Chapter 6)	to be reviewed	availability	information needs	gaps		
	Bias	Results of the	There are several	Criticality	None identified. The		
	determination	benchmark	guidance documents	benchmarking for	review method calls		
		calculations and	on benchmarking	fresh TRISO fuel with	for evaluating		
		bias evaluations	criticality evaluations	higher enrichment is	whether the applicant		
			(ANS, 2007;	to be evaluated,	demonstrates that		
			NRC, 1997, 2001).	given the potential	the benchmark		
			Criticality	lack of criticality	calculations are		
			benchmarking for fresh	benchmark data	adequately		
			TRISO fuel with higher		converged and		
			enrichment is limited.		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 Cred	lit Evaluation for	Not applicable	Not applicable	Not applicable	Not applicable		
Commercial Light-	-Water Reactor	because of					
Spent Nuclear Fue		unirradiated fuel					
			(ANOL(ANO) 0.4.4000 (50007)	"NI I O'II II O (I · ·			
ANS. American Natio	nal Standards Institute	American Nuclear Society	/ (ANSI/ANS) 8.1-1998 (R2007) perican Nuclear Society - 2007	. "Nuclear Criticality Safety In	Operations with		
Jarrell, J. "A Proposed	d Path Forward for Tra	insportation of High-Assav	Low-Enriched Uranium." INL/E	XT-18-51518, Idaho Falls, Id	aho: 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	3, "Directory of Certifica	ates of Compliance for Rac	dioactive Materials Packages, C	ertificates of Compliance." Vo	olume 2, Revision 28.		
NURFG/CR-6	698 "Guide for Validat	tion of Nuclear Criticality S	afety Calculational Methodology	/ " Oak Ridge Tennessee [.] S	cience Applications		
International Corporati	ion. U.S. Nuclear Reg	ulatory Commission. 2001	l.				
NUREG/CR-5	661, "Recommendatio	ns for Preparing the Critica	ality Safety Evaluation of Transp	oortation Packages." ORNL/T	M-11936. Oak Ridge, TN:		
Oak Ridge National La	aboratory. U.S. Nuclea	ar Regulatory Commission	. 1997.				

Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation					
р	ackages for fresh TR	ISO fuel	1	1	
Areas of review		Key information	Information	Potential	Potential guidance
(NUREG-2	216, Chapter 7)	to be reviewed	availability	information needs	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						
Area	s of review	Key information	Information	Potential	Potential guidance	
(NUREG-2	2216, Chapter 7)	to be reviewed	availability	information needs	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					
r	backages for fresh TR	ISO fuel			
Area	s of review	Key information	Information	Potential	Potential guidance
(NUREG-2	2216, Chapter 7)	to be reviewed	availability	information needs	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					
F	oackages for fresh TR	ISO fuel			
Area	s of review	Key information	Information	Potential	Potential guidance
(NUREG-2	2216, Chapter 7)	to be reviewed	availability	information needs	gaps
7.4.4	Tensile properties	Acceptability of	Mechanical properties	None expected	None identified;
Mechanical		material tensile	for commonly used		General acceptance
Properties		properties	packaging materials		criteria not impacted
			are available		by specifics of
					technologies. It is
					assumed that
					commonly used
					packaging materials
					may be also
					adequate for
					package design and
					tabrication for
					fuel. If alternative or
					new materials were
					required in the design
					nackages the SPD
					calls for examination
					of the adequacy of
					information in the
					application related to
					mechanical
					properties of those
					alternative materials
	Fracture resistance	Acceptability of	Mechanical properties	None expected	None identified
		material fracture	for commonly used		General acceptance
		toughness	packaging materials		criteria not impacted
			are available		by specifics of
					technologies

Table I-12.	Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation				
p	ackages for fresh TR	ISO fuel		1	
Areas	s of review	Key information	Information	Potential	Potential guidance
(NUREG-2	216, Chapter 7)	to be reviewed	availability	information needs	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 P	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 Packages for fresh TRISO fuel					
Areas	s of review	Key information	Information	Potential	Potential guidance
(NUREG-2	216, Chapter 7)	to be reviewed	availability	information needs	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 (NUREG-2	s of review 216, 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. II	Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for fresh TRISO fuel					
Areas	s of review	Key information	Information	Potential	Potential guidance	
(NUREG-2	216, Chapter 7)	to be reviewed	availability	information needs	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.	Table I-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation					
p	ackages for fresh TF	RISO fuel		1	1	
Area	s of review	Key information	Information	Potential	Potential guidance	
(NUREG-2	216, Chapter 7)	to be reviewed	availability	information needs	gaps	
		the content	2009, 2010); Detailed		by specifics of	
		materials	evaluation of package		technologies	
			components and fuels			
			is expected to be			
			provided by the			
			applicant			
	Content chemical	Effects of chemical	Detailed evaluation of	None expected	Additional guidance	
	reactions,	reactions,	package components		may need to be	
	outgassing, and	outgassing, and	and fuels is expected		developed to address	
	corrosion	corrosion among	to be provided by the		corrosion of non-fuel	
		the contents and	applicant			
		between the			SPD colle for	
					over calls for	
		components			corrosion wastage	
		components			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 TF	RISO fuel				
Areas of review	Key information	Information	Potential	Potential guidance	
(NUREG-2216, Chapter 7)	to be reviewed	availability	information needs	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 Packages for fresh TRISO fuel					
Areas of review	Key information	Information	Potential	Potential guidance	
(NUREG-2216, Chapter 7)	to be reviewed	availability	information needs	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	s of review	Key information	Information	Potential	Potential guidance	
(NUREG-2	216, Chapter 7)	to be reviewed	availability	information needs	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 N	laterial	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 revie backages for fresh TR	ewed and potential g ISO fuel	gaps for evaluating mate	erials performance of	transportation	
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Area (NUREG-2	s of review 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-03 ML13309A031. W	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					
fac	ilities for spent m	etal fuel	1	1	
Areas of	f review	Key information	Information	Potential	Potential guidance
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	gaps
4.5.1 Description	of the Structures, S	Systems, and Compor	nents		
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 o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	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 Stru and Components Approval	ctures, Systems, Subject to NRC	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 Crite	eria						
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 o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	gaps		
	Fuel and	Fuel rod and	Detailed design	None expected	None identified;		
	cladding	cladding design	information is expected		General acceptance		
		criteria	to be provided by the		criteria not impacted		
			applicant		by specifics of		
					technologies		
	Transfer cask	Transfer cask	Detailed design	None expected	None identified;		
		design criteria	information is expected		General acceptance		
			to be provided by the		criteria not impacted		
			applicant		by specifics of		
					technologies		
	Storage	Storage overpack	Detailed design	None expected	None identified;		
	overpack	design criteria	information is expected		General acceptance		
	(horizontal,		to be provided by the		criteria not impacted		
	vertical,		applicant		by specifics of		
	underground)				technologies		
	Independent	ISFSI concrete	Detailed design	None expected	None identified;		
	spent fuel	pad design criteria	information is expected		General acceptance		
	storage		to be provided by the		criteria not impacted		
	installations		applicant		by specifics of		
	concrete pad				technologies		
4.5.2.2 Other Stru	ictures, Systems,	Design	Detailed design	None expected	None identified;		
and Components	Subject to NRC	descriptions of	information is expected		General acceptance		
Approval		other SSCs	to be provided by the		criteria not impacted		
		subject to NRC	applicant		by specifics of		
	-	approval			technologies		
4.5.3 Loads	Normal	Loads during	Structural performance	None expected	None identified;		
	conditions	normal conditions	of the storage container		General acceptance		
			under load conditions is		criteria not impacted		
			expected to be provided		by specifics of		
			by the applicant		technologies		
	Off-normal	Loads during off-	Structural performance	None expected	None identified;		
	conditions	normal conditions	of the storage container		General acceptance		

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and								
fac	facilities for spent metal fuel							
Areas of	f review	Key information	Information	Potential	Potential guidance			
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	gaps			
			under load conditions is		criteria not impacted			
			expected to be provided		by specifics of			
			by the applicant		technologies			
	Accident conditio	ns						
	Cask drop and	Load conditions	Structural performance	None expected	None identified;			
	tipover	associated with	of the storage container		General acceptance			
		cask drop and	under load conditions is		criteria not impacted			
		tipover	expected to be provided		by specifics of			
			by the applicant		technologies			
	Earthquake	Load conditions	Structural performance	None expected	None identified;			
		associated with	of the storage container		General acceptance			
		earthquake	under load conditions is		criteria not impacted			
			expected to be provided		by specifics of			
			by the applicant		technologies			
	Tornado winds	Load conditions	Structural performance	None expected	None identified;			
		associated with	of the storage container		General acceptance			
		tornado winds	under load conditions is		criteria not impacted			
			expected to be provided		by specifics of			
			by the applicant		technologies			
	Tornado	Load conditions	Structural performance	None expected	None identified;			
	missiles	associated with	of the storage container		General acceptance			
		tornado missiles	under load conditions is		criteria not impacted			
			expected to be provided		by specifics of			
			by the applicant		technologies			
	Flood	Load conditions	Structural performance	None expected	None identified;			
		associated with	of the storage container		General acceptance			
		flood	under load conditions is		criteria not impacted			
			expected to be provided		by specifics of			
			by the applicant		technologies			
	Fire	Load conditions	Structural performance	None expected	None identified;			
		associated with	of the storage container		General acceptance			
		fire	under load conditions is		criteria not impacted			

Table II-1. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and						
fac	ilities for spent m	etal fuel	lufe mus sti s s	Deterstial	Detential milder	
	T review	key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	gaps	
			expected to be provided		by specifics of	
			by the applicant		technologies	
	Explosive	Load conditions	Structural performance	None expected	None identified;	
	overpressure	associated with	of the storage container		General acceptance	
		explosive	under load conditions is		criteria not impacted	
		overpressure	expected to be provided		by specifics of	
			by the applicant		technologies	
4.5.4 Analytical	Hand	Structural analysis	Structural analysis of	None expected	None identified;	
Approach	calculations	of various loading	various loading		General acceptance	
		combinations	combinations is		criteria not impacted	
		using hand	expected to be provided		by specifics of	
		calculations	by the applicant		technologies	
	Finite element	Structural analysis	Structural analysis of	None expected	None identified;	
	analyses	of various loading	various loading		General acceptance	
		combinations	combinations is		criteria not impacted	
		using finite	expected to be provided		by specifics of	
		element analyses	by the applicant		technologies	
4.5.5 Normal and	Off-Normal Conditi	ons				
4.5.5.1	Canister and	Structural analysis	Structural analysis of	None expected	None identified;	
Structures,	associated	of canister and	various loading		General acceptance	
Systems, and	welds and bolts	associated welds	combinations is		criteria not impacted	
Components		and bolts for	expected to be provided		by specifics of	
Important to		various loading	by the applicant		technologies	
Safety		combinations	,		Ũ	
-	Fuel basket	Structural analysis	Structural analysis of	None expected	None identified;	
		of fuel basket and	various loading		General acceptance	
		associated welds	combinations is		criteria not impacted	
		for various loading	expected to be provided		by specifics of	
		combinations	by the applicant		technologies	
	Spent fuel	Structural analysis	Structural analysis of	None expected	None identified:	
	assemblies and	of spent fuel	various loading		General acceptance	
	cladding	assemblies and	combinations is		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 o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	gaps		
		cladding for	expected to be provided		by specifics of		
		various loading	by the applicant		technologies		
		combinations					
	Transfer cask	Structural analysis	Structural analysis of	None expected	None identified;		
		of transfer cask	various loading		General acceptance		
		components for	combinations is		criteria not impacted		
		various loading	expected to be provided		by specifics of		
		combinations	by the applicant		technologies		
	Storage	Structural analysis	Structural analysis of	None expected	None identified;		
	overpack	of steel and	various loading		General acceptance		
		reinforced	combinations is		criteria not impacted		
		concrete	expected to be provided		by specifics of		
		structures for	by the applicant		technologies		
		various loading					
		combinations					
4.5.5.2 Other Stru	ctures, Systems,	Structural analysis	Structural analysis of	None expected	None identified;		
and Components	Subject to NRC	of other SSCs	various loading		General acceptance		
Approval		subject to NRC	combinations is		criteria not impacted		
		approval for	expected to be provided		by specifics of		
		various loading	by the applicant		technologies		
	u diti a u a	complinations					
4.5.6 Accident Co	naitions						
4.5.6.1	Canister and	Structural analysis	Structural analysis of	None expected	None identified;		
Structures,	associated	of canister and	various loading		General acceptance		
Systems, and	welds and bolts	associated welds	combinations is		criteria not impacted		
Components		and bolts for	expected to be provided		by specifics of		
Important to		various loading	by the applicant		technologies		
Salety	First basket		Otwystymal an alysia of		Nexe identified.		
	Fuel basket	Structural analysis	Structural analysis of	None expected	None identified;		
		or ruer basket and			General acceptance		
		associated welds	complinations is		cilieria 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 o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	gaps		
		for various loading	expected to be provided		by specifics of		
		combinations	by the applicant		technologies		
	Spent fuel	Structural analysis	Structural analysis of	None expected	None identified;		
	assemblies and	of spent fuel	various loading		General acceptance		
	cladding	assemblies and	combinations is		criteria not impacted		
		cladding for	expected to be provided		by specifics of		
		various loading	by the applicant		technologies		
		combinations					
	Transfer cask	Structural analysis	Structural analysis of	None expected	None identified;		
		of transfer cask	various loading		General acceptance		
		components for	combinations is		criteria not impacted		
		various loading	expected to be provided		by specifics of		
		combinations	by the applicant		technologies		
	Storage	Structural analysis	Structural analysis of	None expected	None identified;		
	overpack	of steel and	various loading		General acceptance		
		reinforced	combinations is		criteria not impacted		
		concrete	expected to be provided		by specifics of		
		structures for	by the applicant		technologies		
		various loading					
		combinations					
4.5.6.2 Other Stru	ctures, Systems,	Structural analysis	Structural analysis of	None expected	None identified;		
and Components		of other SSCs	various loading		General acceptance		
		subject to NRC	combinations is		criteria not impacted		
		approval for	expected to be provided		by specifics of		
		various loading	by the applicant		technologies		
		combinations					
INL. "Idaho National L	aboratory Preferred Di	sposition Plan for Sodium-	Bonded Spent Nuclear Fuel." Ida	ho Falls, Idaho: Idaho Nation	al Laboratory. 2007.		

Table II-2. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems								
and	and facilities for spent metal fuel							
Areas of	review	Key information	Information	Potential	Potential guidance			
(NUREG-2215	5, Chapter 5)	to be reviewed	availability	information needs	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. Info	Ie II-2. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems							
and	and facilities for spent metal fuel							
Areas of	review	Key information to be reviewed	Information	Potential	Potential guidance			
(NUREG-2215	5, Chapter 5)		availability	information needs	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. Info	Cable II-2. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems							
anc	and facilities for spent metal fuel							
Areas of	review	Key information	Information	Potential	Potential guidance			
(NUREG-221	5, Chapter 5)	to be reviewed	availability	information needs	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			

Areas of review (NUREG-2215, Chapter 5) Key information to be reviewed Information availability Potential information needs Potential gaps Computer codes Computer codes Computer codes used for thermal analysis Codes used in the expected to be available None expected None identified; General acceptance criteria not impacted by specifics of technologies Temperature calculations Maximum and minimum calculations Maximum and minimum expected to be available 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 accident confirmatory analysis Detailed thermal evaluation is expected in the pressure analysis None 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 requirements 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 5.5.5 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 5.5.5 Surveill	Table II-2.Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems							
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(NUREG-2715, Chapter 3) to be reviewed availability Internation needs gaps Computer Computer codes used for thermal analysis Computer codes 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 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 Detailed thermal ev	Areas of		Key information	Information	Potential	Potential guidance		
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Image: section of the section of th		codes	used for thermal	thermal analysis are		General acceptance		
Image: Second state Available Dy specifics of technologies Temperature calculations Maximum and minimum tealculations 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 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 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 evaluation is expected to be provided by the applicant None expected by the applicant None identified; Gener			anaiysis	expected to be		criteria not impacted		
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5.5.5 Surveillance Requirements Surveillance requirements related to thermal evaluations Detailed thermal evaluation is expected to be provided by the applicant None expected to be provided by the applicant 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.			analysis	to be provided by the		criteria not impacted		
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5.5.5 Surveillance Requirements Surveillance Detailed thermal None expected None identified; requirements requirements evaluation is expected to be provided by the order thermal General acceptance valuations applicant applicant 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.			-			technologies		
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related to thermal evaluations to be provided by the applicant 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-10Zr alloys: A critical Review." Nuclear Technology. Vol. 203. pp.109–128. 2018.			requirements	evaluation is expected		General acceptance		
evaluations applicant 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-10Zr alloys: A critical Review." Nuclear Technology. Vol. 203. pp.109–128. 2018.			related to thermal	to be provided by the		criteria not impacted		
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.			evaluations	applicant		by specifics of		
Janney, D.E. and S.L. Hayes. Experimentally Known Fropenies of 0-1021 alloys. A Childai Neview. Nuclear rechnology. Vol. 203. pp.109–120. 2010.	Inney, D.E. and S.L. Havea, "Experimentally Known Properties of U 107r alloys: A critical Paview", Nuclear Technologies, Vol. 202, pp. 100, 129, 2019							
Janney, D.L., J.L. hayes, and J.A. Aukins. A United Neview of the Experimentally Known hopefules of U-1 u-21 Alloys. Fait 1. Thases and thase	Janney, D.E., S.L. Hav	es, and C.A. Adkins.	"A Critical Review of the E	xperimentally Known Properties	of U-Pu-Zr Allovs. Part 1: F	Phases and Phase		
Diagrams." Nuclear Technology. Vol. 205. pp.1,387–1,415. 2019.	Diagrams." Nuclear Te	echnology. Vol. 205.	pp.1,387–1,415. 2019.	. , , ,	2			
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	Janney, D.E., S.L. Hay	es, and C.A. Adkins.	"A Critical Review of the E	xperimentally Known Properties	of U-Pu-Zr Alloys. Part 2:	Thermal and Mechanical		

Table II-2. Information to be rev	viewed and potential	gaps for evaluating them	mal performance of d	Iry storage systems				
and facilities for spe	and facilities for spent metal fuel							
Areas of review	Key information	Information	Potential	Potential guidance				
(NUREG-2215, Chapter 5)	to be reviewed	availability	information needs	gaps				
Janney, D.E. "Metallic Fuels Handbook, Part	1: Alloys Based on U-Zr,	Pu-Zr, U-Pu, or U-Pu-Zr, Includi	ng Those with Minor Actinid	es (Np, Am, Cm), Rare-				
earth Elements (La, Ce, Pr, Nd, Gd), and Y."	INL/EXT-15-36520 Revisi	on 3 Part 1. Idaho Falls, Idaho:	Idaho National Laboratory.	2018a.				
"Metallic Fuels Handbook, Part 2: Ele	ements and Alloys not Base	ed on U-Zr, Pu-Zr, U-Pu, or U-Pι	u-Zr." INL/EXT-15-36520 Re	evision 3 Part 2. Idaho Falls,				
Idaho: Idaho National Laboratory. 2018b.	Idaho: Idaho National Laboratory. 2018b.							
Leibowitz, L., E.C. Chang, M.G. Chasanov, R	R.L. Gibby, C. Kim, A.C. Mil	llunzi, D. Stahl. "Properties for Li	quid Metal Fast Breeder Re	actor Safety Analysis."				
Argonne National Laboratory. ANL-CEN-RSE	D-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 o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 6)	to be reviewed	availability	information needs	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. Informa	Fable 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		Key information	Information	Potential	Potential guidance		
(NUREG-2215, Ch	apter 6)	to be reviewed	availability	information needs	gaps		
Gam	nma sources	Gamma source	Metal fuel fabricated	Source-term	None identified. The		
		strengths and	with reprocessed	specification	review method calls		
		spectra for the	uranium could possibly	including gamma	for verifying the		
		contents and	contain radioactive	sources and their	gamma source terms		
		activated hardware	sources; Although	energies arising	as a function of		
			ASTM C996 (ASTM,	from fission	energy to ensure that		
			2021) is frequently	products from	the total source is		
			applied, information in	reprocessed	correctly considered in		
			the literature is limited	uranium or other	the shielding		
				impurities is to be	evaluation. The		
				evaluated	existing review		
					method is sufficient to		
					evaluate residual		
					gamma sources in		
					metal fuel fabricated		
					with reprocessed		
			6		uranium.		
Neut	tron sources	Neutron source	Detailed shielding	None expected	None identified;		
		strengths and	evaluation is expected		General acceptance		
		spectra for the	to be provided by the		criteria not impacted		
		contents and	applicant		by specifics of		
		activated hardware		N I (1	technologies		
Othe	er	Information	Detailed shielding	None expected	None identified;		
para	ameters	concerning reactor	evaluation is expected		General acceptance		
affec	cting the	operations that	to be provided by the		criteria not impacted		
sour	rce term	arrect the source	applicant		by specifics of		
		terms for the			technologies		
		shielding analyses					

Table II-3. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems							
an	and facilities for spent metal fuel						
Areas o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 6)	to be reviewed	availability	information needs	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 o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 6)	to be reviewed	availability	information needs	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- Not ap Related GTCC Waste Storage (SL) spent		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" < <u>https://compass.astm.org/EDIT/html_annot.cgi?C996+20</u> > (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	f review	Key information	Information	Potential	Potential guidance
7.5.1 Criticality De Features	sign Criteria and	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	gaps 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. Info	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 <i>k</i> _{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. Inforr syste	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 r	eview	Key information	Information	Potential	Potential guidance		
(NUREG-2215,	Chapter 7)	to be reviewed	availability	information needs	gaps		
E	3enchmark 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. Info	Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for spent metal fuel						
Areas of (NUREG-221	review 5, 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 <i>k</i> 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. Info	Table II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage systems and facilities for sport metal fuel					
	systems and facilities for spent metal fuel Areas of review Retential Retential Retential Retential Retential					
(NUREG-221	5. Chapter 7)	to be reviewed	availability	information needs		
	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						
sys	systems and facilities for spent metal fuel					
Areas of	i review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 7)	to be reviewed	availability	information needs	gaps	
	Code validation— <i>k</i> 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-Rela Than-Class-C Was (SL)	ated Greater- ste and HLW	Not applicable because of irrelevant wastes	Not applicable	Not applicable	Not applicable	
1.5.7 Supplementa	al information	Not applicable	Not applicable	Not applicable	Not applicable	

Table II-4. Information to be rev	le II-4. Information to be reviewed and potential gaps for evaluating criticality performance of dry storage						
systems and facilitie	s for spent metal fu	el					
Areas of review	Key information	Information	Potential	Potential guidance			
(NUREG-2215, Chapter 7)	to be reviewed	availability	information needs	gaps			
ANS. American National Standards Institute/	American Nuclear Society	(ANSI/ANS) 8.1-1998 (R2007).	"Nuclear Criticality Safety in C	Operations with Fissionable			
Materials Outside Reactors." La Grange Parl	k, Illinois: American Nuclea	ar Society. 2007.					
FRWG. "Nuclear Metal Fuel: Characteristics,	Design, Manufacturing, T	esting, and Operating History." V	White Paper 18-01. ML18165/	A249. Fast Reactor Working			
Group. 2018.							
INL. "Idaho National Laboratory Preferred Di	sposition Plan for Sodium-	Bonded Spent Nuclear Fuel." Ida	aho Falls, Idaho: Idaho Natior	nal Laboratory. 2007.			
Jarrell, J. "A Proposed Path Forward for Trar	nsportation of High-Assay I	_ow-Enriched Uranium." INL/EX	T-18-51518. Idaho Falls, Idah	o: Idaho National			
Laboratory. 2018.							
NRC. NUREG/CR-6698, "Guide for Validation	on of Nuclear Criticality Sa	fety Calculational Methodology."	' Oak Ridge, Tennessee: Scie	nce Applications			
International Corporation. U.S. Nuclear Regu	ilatory Commission. 2001.						
NUREG/CR-6361, "Criticality Benchi	mark Guide for Light-Wate	r-Reactor Fuel in Transportation	and Storage Packages." OR	NL/TM-13211. Oak Ridge,			
Tennessee: Oak Ridge National Laboratory.	U.S. Nuclear Regulatory C	Commission. 1997.					
NWTRB. "Management and Disposal of U.S.	Department of Energy Sp	ent Nuclear Fuel." Washington,	DC: Nuclear Waste Technica	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 C	Continued Wet Storage of EBR-I	I Fuel." Reno, Nevada: DOE S	Spent Nuclear Fuel and			
Fissile Material Management Conference. 19	996.						

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 o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	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. Infe	able II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems							
and	and facilities for spent metal fuel							
Areas o	f review	Key information	Information	Potential	Potential guidance			
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	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							
	Areas of review Key information Information Betential Retential guidance						
(NUREG-221	5. Chapter 8)	to be reviewed	availability	information needs	aaps		
	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	l'ensile 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 o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	gaps	
	Fracture	Acceptability of	Mechanical properties	None expected	None identified;	
	resistance	material fracture	for commonly used		General acceptance	
		toughness	storage container		criteria not impacted	
			materials are available		by specifics of	
					technologies	
	Performance of	Acceptability of the	Mechanical properties	None expected	None identified;	
	aluminum	tensile properties,	for commonly used		General acceptance	
	components	fracture toughness,	aluminum materials are		criteria not impacted	
		and creep of	available		by specifics of	
			T I I <i>I</i> C			
8.5.5 Thermal Pro	perties	I nermal properties	I nermal properties for	Data characterizing	None identified. The	
				diograma boot	for vorifying the	
		motorials: Effect of	storage container	capacity and	thermal properties and	
		degradation and	Some thermal	thermal properties	the change in these	
		anisotronic	properties of fuel pin	of metal fuel and	properties from	
		dependencies of	components structural	converted waste	material degradation	
		thermal properties	components, and metal	forms are limited	The existing review	
			fuel are available		method is sufficient to	
			(Leibowitz et al., 1976;		deal with the	
			Janney, 2018a, b;		availability of	
			Janney and Hayes,		information related to	
			2018; Janney et al.,		metal fuel properties	
			2020, 2019)		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 o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	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. Infe	Table II-5. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems							
and	and facilities for spent metal fuel							
Areas o	f review	Key information	Information	Potential	Potential guidance			
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	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 o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	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	and facilities for spent metal fuel							
Areas of	f review	Key information	Information	Potential	Potential guidance			
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	gaps			
	Austenitic stainless steels	Localized corrosion and stress corrosion cracking in chloride- containing environments; Chloride-induced stress corrosion cracking in sensitized stainless	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	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	and facilities for spent metal fuel							
Areas o	f review	Key information	Information	Potential	Potential guidance			
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	gaps			
	Coating	Coating	Detailed evaluation of	None expected	None identified;			
	selection	manufacturer, type	storage container		General acceptance			
		of primers and	materials is expected		criteria not impacted			
		topcoat, coating	to be provided by the		by specifics of			
		thickness, and	applicant		technologies			
		ability of the						
		coating to						
		withstand the						
		in-service						
		conditions						
	Coating	Qualification	Detailed evaluation of	None expected	None identified;			
	qualification	testing for coating	storage container		General acceptance			
	testing	performance in	materials is expected		criteria not impacted			
		accordance with	to be provided by the		by specifics of			
		several standard	applicant		technologies. The			
		ASTM			SRP calls for			
		(and possibly			evaluating any			
		other) tests			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 spont metal fuel							
Areas of review Key information Information Potential Potential quidan							
(NUREG-2215, Chapter 8)		to be reviewed	availability	information needs	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							
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Areas of	Areas of review Key information Information Potential Potential guidance						
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance	
(NUREG-2215, Chapter 8)	to be reviewed	availability	information needs	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 Alexander, D.J. and R.K. Nanstand. "The E Proceedings of the Seventh International Systems	Performance of the fuel can for damaged fuel fects of Aging for 50,000 Ho mposium on Environmental	Detailed evaluation of the fuel can performance is expected to be provided by the applicant ours at 343°C on the Mechanical Degradation of Materials in Nuc	None expected Properties of Type 308 Sta	None identified; General acceptance criteria not impacted by specifics of technologies inless Steel Weldments." r Reactors. Breckenridge,	
Colorado. NACE. Houston, Texas. pp. 747-	758.1995. Never D. Burkes C. Lee T.	Mizuno F Delage and L Some	ers "Metallic Fuels for Adv	anced Reactors " .lournal of	

Nuclear Materials. Vol. 392. pp. 139–150. 2009.

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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.

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Table II-5.Information to be rev	iewed and potential	gaps for evaluating mate	erials performance of	dry storage systems
and facilities for sper	nt metal fuel			
Areas of review	Key information	Information	Potential	Potential guidance
(NUREG-2215, Chapter 8)	to be reviewed	availability	information needs	gaps
Janney, D.E. "Metallic Fuels Handbook, Part	1: Alloys Based on U-Zr, I	Pu-Zr, U-Pu, or U-Pu-Zr, Includir	ng Those with Minor Actinid	es (Np, Am, Cm), Rare-
earth Elements (La, Ce, Pr, Nd, Gd), and Y." I	NL/EXT-15-36520 Revisio	n 3 Part 1. Idaho Falls, Idaho:	Idaho National Laboratory.	2018a.
"Metallic Fuels Handbook, Part 2: Ele	ments and Alloys not Base	ed on U-Zr, Pu-Zr, U-Pu, or U-Pi	u-Zr." INL/EXT-15-36520 R	evision 3 Part 2. Idaho
Falls, Idaho: Idaho National Laboratory. 2018	3b.			
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Diagrams." Nuclear Technology. Vol. 205. p	p.1,387–1,415. 2019.			
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Properties." Nuclear Technology. Vol. 206. p	p.1–22. 2020.			
Leibowitz, L., E.C. Chang, M.G. Chasanov, R.	L. Gibby, C. Kim, A.C. Mill	unzi, and D. Stahl. "Properties for	or Liquid Metal Fast Breede	r Reactor Safety Analysis."
Diagrams." <i>Nuclear Technology</i> . Vol. 205. p Janney, D.E., S.L. Hayes, and C.A. Adkins. " Properties." <i>Nuclear Technology</i> . Vol. 206. p Leibowitz, L., E.C. Chang, M.G. Chasanov, R.	p.1,387–1,415. 2019. A Critical Review of the Ex pp.1–22. 2020. L. Gibby, C. Kim, A.C. Mill	perimentally Known Properties o unzi, and D. Stahl. "Properties fo	of U-Pu-Zr Alloys. Part 2: 1 or Liquid Metal Fast Breede	Thermal and Mechanical r 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 Key information Information Potential Potential guidance						
(NUREG-2215	5, Chapter 9)	to be reviewed	availability	information needs	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 Release	n Potential for	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		Key information	Information	Potential	Potential guidance
(NUREG-2215	, Chapter 9)	to be reviewed	availability	information needs	gaps
9.5.4 Confinement Analyses	Normal conditions	Confinement analysis and the resulting doses for	Detailed confinement evaluation is expected to be provided by the	None expected	None identified; General acceptance criteria not impacted
		the normal conditions	applicant		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 t	Table II-6. Information to be reviewed and potential gaps for evaluating confinement performance of dry storage systems and facilities for spont metal fuel					
Areas of review (NUREG-2215, Chapter	(9) to be reviewed	Information availability	Potential information needs	Potential guidance		
Evaluatio release estimate: spent nu fuel and level radioactiv waste (S	on of Dose calculations and release s for estimates for clear normal operations, high-off-normal operations, and ve design-basis L) 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 release estimates reactor-ro- greater the Class Co- (SL)	on of Not applicable to spent metal fuel s for elated han waste	Not applicable	Not applicable	Not applicable		
9.5.5 Supplemental Informat	tion 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. Info	-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and facilities for spont TRISO fuel				
Areas of	f review	Key information	Information	Potential	Potential guidance
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	gaps
4.5.1 Description	of the Structures, S	Systems, and Compor	nents		
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. Inf	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 Key information Information Potential Potential guidar						
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	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 Crite	eria					
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					
	f roviow	KISU fuel	Information	Potential	Potential guidance
(NUREG-221	5 Chanter 4)	to be reviewed	availability	information needs	rotential guidance
	Fuel and	Fuel rod and	Detailed design	None expected	None identified:
	cladding	cladding design	information is expected		General accentance
	oladanığ	criteria	to be provided by the		criteria not impacted
			applicant		by specifics of
					technologies
	Transfer cask	Transfer cask	Detailed design	None expected	None identified;
		design criteria	information is expected	·	General acceptance
			to be provided by the		criteria not impacted
			applicant		by specifics of
					technologies
	Storage	Storage overpack	Detailed design	None expected	None identified;
	overpack	design criteria	information is expected		General acceptance
	(horizontal,		to be provided by the		criteria not impacted
	vertical,		applicant		by specifics of
	underground)				technologies
	Independent	ISFSI concrete	Detailed design	None expected	None identified;
	spent fuel	pad design criteria	information is expected		General acceptance
	storage		to be provided by the		criteria not impacted
	installations		applicant		by specifics of
	concrete pad				technologies
4.5.2.2 Other Stru	ctures, Systems,	Design	Detailed design	None expected	None identified;
and Components	Subject to NRC	descriptions of	information is expected		General acceptance
Approval		other SSCs	to be provided by the		criteria not impacted
		subject to NRC	applicant		by specifics of
		approval			technologies
4.5.3 Loads	Normal	Loads during	Structural performance	None expected	None identified;
	conditions	normal conditions	of the storage container		General acceptance
			under load conditions is		criteria not impacted
			expected to be provided		by specifics of
	0.55	l de - de mine en	by the applicant	Niewe erweiten	technologies
	Off-normal	Loads during off-	Structural performance	None expected	None identified;
	conditions	normal conditions	of the storage container		General acceptance

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and					
fac	ilities for spent T	RISO fuel		D. G. C.L	
Areas of	review	Key information	Information	Potential	Potential guidance
(NUREG-221	5, Chapter 4)	to be reviewed	availability	Information needs	gaps
			under load conditions is		criteria not impacted
			expected to be provided		by specifics of
			by the applicant		technologies
	Accident conditio	ns			
	Cask drop and	Load conditions	Structural performance	None expected	None identified;
	tipover	associated with	of the storage container		General acceptance
		cask drop and	under load conditions is		criteria not impacted
		tipover	expected to be provided		by specifics of
			by the applicant		technologies
	Earthquake	Load conditions	Structural performance	None expected	None identified;
		associated with	of the storage container		General acceptance
		earthquake	under load conditions is		criteria not impacted
			expected to be provided		by specifics of
			by the applicant		technologies
	Tornado winds	Load conditions	Structural performance	None expected	None identified;
		associated with	of the storage container		General acceptance
		tornado winds	under load conditions is		criteria not impacted
			expected to be provided		by specifics of
			by the applicant		technologies
	Tornado	Load conditions	Structural performance	None expected	None identified;
	missiles	associated with	of the storage container	•	General acceptance
		tornado missiles	under load conditions is		criteria not impacted
			expected to be provided		by specifics of
			by the applicant		technologies
	Flood	Load conditions	Structural performance	None expected	None identified:
		associated with	of the storage container		General acceptance
		flood	under load conditions is		criteria not impacted
			expected to be provided		by specifics of
			by the applicant		technologies
	Fire	Load conditions	Structural performance	None expected	None identified
		associated with	of the storage container		General accentance
		fire	under load conditions is		criteria not impacted

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and					
Areas of	f review	Key information	Information	Potential	Potential guidance
			expected to be provided by the applicant	momation needs	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 Conditi	ons	· · ·		
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					
fac	ilities for spent T	RISO fuel	la fa mu a ti a n	Detential	Detential muidemen
	review	Key information	Information	Potential	Potential guidance
(NUREG-221	5, Chapter 4)	to be reviewed	availability	Information needs	gaps
		cladding for	expected to be provided		by specifics of
		various loading	by the applicant		technologies
		combinations			
	I ransfer cask	Structural analysis	Structural analysis of	None expected	None identified;
		of transfer cask	various loading		General acceptance
		components for	combinations is		criteria not impacted
		various loading	expected to be provided		by specifics of
		combinations	by the applicant		technologies
	Storage	Structural analysis	Structural analysis of	None expected	None identified;
	overpack	of steel and	various loading		General acceptance
		reinforced	combinations is		criteria not impacted
		concrete	expected to be provided		by specifics of
		structures for	by the applicant		technologies
		various loading			
		combinations			
4.5.5.2 Other Stru	ctures, Systems,	Structural analysis	Structural analysis of	None expected	None identified;
and Components	Subject to INRC	of other SSCs	various loading		General acceptance
Approvai					criteria not impacted
		approval for	expected to be provided		by specifics of
		various loading	by the applicant		technologies
A E C Assident Ca		complinations			
4.5.6 Accident Co	naitions				
4.5.6.1	Canister and	Structural analysis	Structural analysis of	None expected	None identified;
Structures,	associated	of canister and	various loading		General acceptance
Systems, and	welds and bolts	associated welds	combinations is		criteria not impacted
Components		and bolts for	expected to be provided		by specifics of
Important to		various loading	by the applicant		technologies
Safety		combinations			
	Fuel basket	Structural analysis	Structural analysis of	None expected	None identified;
		of fuel basket and	various loading		General acceptance
		associated welds	combinations is		criteria not impacted

Table II-7. Information to be reviewed and potential gaps for evaluating structural integrity of dry storage systems and forsitivities for event TPICO fuel						
Areas of	f review	Kiso fuel Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 4)	to be reviewed	availability	information needs	gaps	
	, <u> </u>	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 Stru and Components	ctures, Systems,	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 Department of Energy NRC. "Safety Evaluati Nuclear Regulatory Co	Combinations Image: Combination independent spect of the second seco					

Table II-8. Info and	ble 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 (NUREG-2215	review 5, 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						
	roviow	Kov information	Information	Potential	Potontial guidanco	
(NUREG-221	5 Chapter 5)	to be reviewed	availability	information needs	rolential guidance	
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 sport TPISO fuel							
Areas of	Areas of reviewKey informationInformationPotential						
(NUREG-221	5, Chapter 5)	to be reviewed	availability	information needs	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. Info	Table II-8. Information to be reviewed and potential gaps for evaluating thermal performance of dry storage systems and facilities for sport TPISO fuel						
Areas of (NUREG-221	f review 5. 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								
an	and facilities for spent TRISO fuel							
Areas o	f review	Key information	Information	Potential	Potential guidance			
(NUREG-221	5, Chapter 6)	to be reviewed	availability	information needs	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. Info	Table II-9. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems							
	Areas of review Key information Information Detential Potential suidence							
	Chapter 6)	to be reviewed	availability	information noods				
(NUREG-2215	5, Chapter 6) Gamma sources	to be reviewed Gamma source strengths and spectra for the contents and activated hardware	availability 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	information needs Source-term specification including gamma sources and their energies arising from fission products from reprocessed uranium or other impurities is to be evaluated	gaps 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 o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 6)	to be reviewed	availability	information needs	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. Inf	ble II-9. Information to be reviewed and potential gaps for evaluating shielding performance of dry storage systems								
an	and facilities for spent TRISO fuel								
Areas o	f review	Key information	Information	Potential	Potential guidance				
(NUREG-221	5, Chapter 6)	to be reviewed	availability	information needs	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				
	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- Not applicat Related GTCC Waste Storage (SL) spent TRISC		Not applicable to spent TRISO fuel	Not applicable	Not applicable	Not applicable				
ASTM. C996. "Standa	ard Specification for Ura	anium Hexafluoride Enrich	ed to Less Than 5 percent ²³⁵ U"						

<<u>https://compass.astm.org/EDIT/html_annot.cgi?C996+20</u>> (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. Info	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		Key information	Information	Potential	Potential guidance		
7.5.1 Criticality De Features	sign Criteria and	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 o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 7)	to be reviewed	availability	information needs	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		Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 7)	to be reviewed	availability	information needs	gaps		
	Neutron multiplication factor	Results of the <i>k</i> _{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. Info sys	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	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	5, Chapter 7)	to be reviewed	availability	information needs	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 <i>k</i> 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	Key information	Information	Potential	Potential guidance	
	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					
sys	tems and facilitie	s for spent TRISO fu	el		
Areas of review		Key information	Information	Potential	Potential guidance
(NUREG-2215, Chapter 7)		to be reviewed	availability	information needs	gaps
	Code validation—keff	Bias and bias	There are several quidance documents	Criticality benchmarking for	Additional guidance
	determination	associated with actinide-only, and fission product and minor actinide burnup credit	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).	spent TRISO fuel with enrichments between 5 and 20 weight percent ²³⁵ U is to be evaluated	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-Rela Than-Class-C Wa (SL)	ated Greater- ste and HLW	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	Key information	Information	Potential	Potential guidance
(NUREG-2215, Chapter 7)	to be reviewed	availability	information needs	gaps
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 o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	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. Info	able II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems and facilities for spent TRISO fuel					
Areas o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	gaps	
	Confinement	Weld inspection	Detailed weld	None expected	None identified;	
	weld inspection	methods and	inspection information		General acceptance	
		requirements	is expected to be		criteria not impacted	
			provided by the		by specifics of	
			applicant		technologies	
	Confinement	Weld testing	Detailed weld testing	None expected	None identified;	
	weld testing	methods and	information is expected		General acceptance	
		requirements	to be provided by the		criteria not impacted	
			applicant		by specifics of	
					technologies	
8.5.4	Tensile	Acceptability of	Mechanical properties	None expected	None identified;	
Mechanical	properties	material tensile	for commonly used		General acceptance	
Properties of		properties	storage container		criteria not impacted	
Metals			materials are available		by specifics of	
	Fracture	Acceptability of	Machanical properties	None expected	Nono identified:	
	raciure	Acceptability of	for commonly used	None expected	Conoral accontance	
	resistance		storage container		critoria not impacted	
		lougimess	materials are available		by specifics of	
					technologies	
	Performance of	Acceptability of the	Mechanical properties	None expected	None identified	
	aluminum	tensile properties	for commonly used		General acceptance	
	components	fracture toughness	aluminum materials are		criteria not impacted	
		and creep of	available		by specifics of	
		aluminum materials			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 Pro	perties	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	d facilities for spe	nt TRISO fuel		1	
Areas of review		Key information	Information	Potential	Potential guidance
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	gaps
	Qualifying the	Qualification of	Detailed evaluation of	None expected	None identified;
	neutron	absorber material	storage container		General acceptance
	absorber	properties not	materials is expected		criteria not impacted
	material	associated with	to be provided by the		by specifics of
	fabrication	neutron attenuation	applicant		technologies
0.5.0.0.1	process				
8.5.8 Concrete	Embedment	Material used for	Detailed design	None expected	None identified;
and Reinforcing	materials	empedments,	Information is expected		General acceptance
Steel		inserts, conduits,	to be provided by the		criteria not impacted
		items embedded in	applicant		by specifics of
		the concrete			lechnologies
	Concrete design		Detailed design	None expected	None identified:
	and temperature	material	information is expected	None expected	General acceptance
	limits	specifications for	to be provided by the		criteria not impacted
	mmis	the concrete.	applicant		by specifics of
		Temperature			technologies
		requirements.			teermenegiee
		Changes in			
		concrete properties			
		with time			
	Omission of	Omission of	Detailed design	None expected	None identified;
	reinforcement	reinforcing steel in	information is expected		General acceptance
		the concrete	to be provided by the		criteria not impacted
			applicant		by specifics of
					technologies
	Radiation	Radiation effects	Detailed evaluation of	None expected	None identified;
	damage	on concrete	storage container		General acceptance
		properties	materials is expected		criteria not impacted
			to be provided by the		by specifics of
			applicant		technologies

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems					
and	d facilities for spe	nt IRISO fuel	Information	Detential	Detential quidence
	f review	Key information	Information	Potential	Potential guidance
(NUREG-221	5, Chapter 8)	to be reviewed		Information needs	gaps
8.5.9 Bolt Applicat	lions	Material properties	Detailed evaluation of	None expected	
		of the boiling;	storage container		General acceptance
		Effect of corrosion	materials is expected		criteria not impacted
		on the boiling	to be provided by the		by specifics of
		materials; Closure	applicant		lechnologies
0 E 10 Casla	Matallia agala	DOIL SURESSES	Detailed evaluation of	Name extend	None identified:
a.b. IU Seals	Metallic seals	Material properties	Detailed evaluation of	None expected	None identified;
		of metallic seals	storage container		General acceptance
			to be provided by the		by specifics of
			applicant		technologies
	Flastomeric	Material properties	Detailed evaluation of	None expected	None identified:
	seals	of elastomeric	storage container		General accentance
	50015	seals. Effects of	materials is expected		criteria not impacted
		thermal radiation	to be provided by the		by specifics of
		and chemical	applicant		technologies
		reactions on the			
		seal materials			
8.5.11 Corrosion	Environments	Range of	Detailed design	None expected	None identified;
Resistance		environmental	information is expected		General acceptance
		conditions	to be provided by the		criteria not impacted
		encountered for	applicant		by specifics of
		storage container			technologies
		components			•
	Carbon and low-	Environment	Detailed evaluation of	None expected	None identified;
	alloy steels	dependencies of	storage container		General acceptance
		corrosion rate;	materials is expected		criteria not impacted
		Coatings for	to be provided by the		by specifics of
		corrosion	applicant		technologies
		prevention			

Table II-11. Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems					
and	d facilities for spe	nt TRISO fuel			
Areas of review		Key information	Information	Potential	Potential guidance
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	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. Info	Information to be reviewed and potential gaps for evaluating materials performance of dry storage systems						
and facilities for spent TRISO fuel							
Areas of review		Key information	Information	Potential	Potential guidance		
(NUREG-2215, Chapter 8)		to be reviewed	availability	information needs	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							
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	Areas of review Key information Information Betential Betential guidance						
(NUREG-221	5. Chapter 8)	to be reviewed	availability	information needs	aaps		
	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	storage container materials is expected	None expected	General acceptance		

Table II-11. Info	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 (NUREG-221)	f review 5, 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 spont TRISO fuel						
Areas of	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	5, Chapter 8)	to be reviewed	availability	information needs	gaps	
			provided by the		by specifics of	
			applicant		technologies	
					J	
DOE. "Safety Analysis	Report for Fort St. Vra	in Independent Spent Fuel	Storage Installation." Revision 8	8. Chapter 3. ML103640368	3. Idaho Falls, Idaho: U.S.	
Department of Energy	Idaho Operations Offic	e. 2010.				
"Characteristic:	s of Potential Repositor	y Wastes." DOE/RW-0184	-R1. Vol 2. ORNL/-2217, Vol 2.	Oak Ridge, Tennessee: Oa	k 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. V	/ol. 191. pp. 113–121. 2	2015.				
Marschman, S.C., F.M	l. Berting, R.G. Clemme	er, E.R. Gilbert, R.J. Guent	her, W.C. Morgan, and P. Sliva.	"Characterization Plan for	Fort St. Vrain and Peach	
Bottom Graphita Eugla	" DNINI 11265 Dichla	nd Machington: Desifie Na	arthweat National Laboratory 10	02		

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	Key information	Information	Potential	Potential guidance		
(NUREG-2215	, Chapter 9)	to be reviewed	availability	information needs	gaps		
9.5.1 Confinement	Design criteria	design criteria	Information available on many NRC-certified	None expected	None identified; General acceptance		
Characteristics			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		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 Capability	Monitoring	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 Release	n Potential for	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								
sys	systems and facilities for spent TRISO fuel							
Areas of	review	Key information	Information	Potential	Potential guidance			
(NUREG-2215	5, Chapter 9)	to be reviewed	availability	information needs	gaps			
9.5.4	Normal	Confinement	Detailed confinement	None expected	None identified;			
Confinement	conditions	analysis and the	evaluation is expected		General acceptance			
Analyses		resulting doses for	to be provided by the		criteria not impacted			
		the normal	applicant		by specifics of			
		conditions			technologies			
	Off-normal	Confinement	Detailed confinement	None expected	None identified;			
	conditions	analysis and the	evaluation is expected		General acceptance			
	(anticipated	resulting doses for	to be provided by the		criteria not impacted			
	occurrences)	the off-normal	applicant		by specifics of			
		conditions			technologies			
	Design-basis	Confinement	Detailed confinement	None expected	None identified;			
	accident	analysis and the	evaluation is expected		General acceptance			
	conditions	resulting doses for	to be provided by the		criteria not impacted			
	(including	the accident	applicant		by specifics of			
	natural	conditions			technologies			
	phenomenon							
	events)							
	Identification of	Spectrum of	Detailed confinement	None expected	None identified;			
	release events	release events for	evaluation is expected		General acceptance			
	(SL)	normal operations,	to be provided by the		criteria not impacted			
		off-normal	applicant		by specifics of			
		operations, and			technologies			
		design-basis						
		accidents						
	Evaluation of	Dose calculations	Detailed confinement	None expected	None identified;			
	release	and release	evaluation is expected		General acceptance			
	estimates for	estimates for	to be provided by the		criteria not impacted			
	spent nuclear	normal operations,	applicant		by specifics of			
	fuel and high-	off-normal			technologies			
	level	operations, and						
	radioactive	design-basis						
	waste (SL)	accidents						

Table II-12. Info	Table II-12. Information to be reviewed and potential gaps for evaluating confinement performance of dry storage							
sys	stems and faciliti	es for spent TRISO f	fuel					
Areas of	review	Key information	Information	Potential	Potential guidance			
(NUREG-221	5, Chapter 9)	to be reviewed	availability	information needs	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 Supplement	al 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. Info for	Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel for spent metal fuel						
Areas of (NUREG-221	f review 6. Chapter 2)	Key information	Information availability	Potential information needs	Potential guidance		
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. Info	Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages						
tor	spent metal fuel	Keesin ferme etien	lu fa mu ati a u	Detential	Detential maidement		
	review	Key information	Information	Potential	Potential guidance		
(NUREG-221)	b, Chapter 2)	to be reviewed		Information needs	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	for spent metal fuel							
Areas of	f review	Key information	Information	Potential	Potential guidance			
(NUREG-221	6, Chapter 2)	to be reviewed	availability	information needs	gaps			
2.4.5 Normal	Heat	Maximum	Structural	None expected	None identified; General			
Conditions of		temperature,	performance of the		acceptance criteria not			
Transport		maximum	package under the		impacted by specifics of			
		pressure, and	heat-loading condition		technologies			
		thermal stress	is expected to be					
		under the heat-	provided by the					
	<u> </u>	loading condition	applicant					
	Cold	Maximum	Structural	None expected	None identified; General			
		temperature,	performance of the		acceptance criteria not			
		minimum	package under the		impacted by specifics of			
		Internal pressure,	cold condition is		technologies			
		and residual stress	expected to be					
			provided by the					
	Deduced			Name expected	None identified: Coneral			
	Reduced		enternel proceure in	None expected	None identined, General			
	brocouro	external pressure	external pressure is		imported by aposition of			
	pressure	external	described and		technologies			
		pressures of the	evaluated by the		teennologies			
		nackade and the	applicant					
		containment						
		system						
	Increased	Effects of	Effects of increased	None expected	None identified: General			
	external	increased	external pressure is		acceptance criteria not			
	pressure	external pressure	expected to be		impacted by specifics of			
	I	on the internal and	described and		technologies			
		external	evaluated by the		3			
		pressures of the	applicant					
		package and the						
		containment						
		system						

Table III-1. Info	able III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages						
for	for spent metal fuel						
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	6, Chapter 2)	to be reviewed	availability	information needs	gaps		
	Vibration and	Effects of vibration	Information available	Sodium creep and	None identified. The		
	fatigue	normally incident	on many NRC-	location shift	review method calls for		
		to transport;	certified package	susceptibility and	evaluating the package		
		Fatigue under the	designs (NRC, 2013);	its effects on the	design for the effects of		
		combined	however, no vibration	geometric	vibration. The existing		
		stresses from	test and analysis of	configuration of	review method is		
		vibration,	packages for	spent metal fuel	sufficient for the		
		temperature, and	transporting sodium-	under the influence	assessment of sodium		
		pressure loads	containing spent metal	of vibration is to be	creep under the		
			fuel are available	evaluated	influence of vibration.		
	Water spray	Effects of the	Detailed water spray	None expected	None identified; General		
		water spray test	test is expected to be		acceptance criteria not		
			provided by the		impacted by specifics of		
			applicant				
	Free drop	Effects of the 0.3	Information available	Sodium creep and	None identified. The		
		to 1.2-m free-drop	on many NRC-		review method calls for		
		test		susceptibility and	evaluating the package		
			designs (INRC, 2013);	its effects on the	free drep. The evicting		
			nowever, no free-drop	geometric configuration of	review method in		
			transporting addium	configuration of	review method is		
			containing sould motal	fuel under the	sufficient for the		
			fuel is available	influence of drop is	creep under the		
				to be evaluated	influence of a drop		
	Corner dron	Not applicable	Not applicable	Not applicable	Not applicable		
		herause of the					
		nackage weight					
		exceedance					
	Compression	Not applicable	Not applicable	Not applicable	Not applicable		
	2 2001011	because of the					
		package weight					
		exceedance					

Table III-1. Info for	Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel							
Areas o (NUREG-221	f review 6, 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. Info	Table III-1. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent metal fuel for spent metal fuel						
Areas of (NUREG-2216	review 5, 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 Conditions for Fiss	Accident ile 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 re	Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages						
for spent metal fuel							
Areas of review	Key information	Information	Potential	Potential guidance			
(NUREG-2216, Chapter 2)	to be reviewed	availability	information needs	gaps			
NRC. NUREG–0383, "Directory of Certifica	ites of Compliance for Radio	oactive Materials Packages, Ce	ertificates of Compliance." `	Volume 2, Revision 28.			
ML13309A031. Washington, DC: U.S. Nuc	clear Regulatory Commissio	on. 2013.					
Oklo Inc. "Part II. Final Safety Analysis Rep	ort." U.S. Nuclear Regulat	ory Commission ADAMS Acces	ssion Number ML20075A00	03. 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	Key information	Information	Potential	Potential guidance		
(NUREG-2216	5, Chapter 3)	to be reviewed	availability	information needs	gaps		
3.4.1 Description	Packaging	Drawings and	Information available on	None expected	None identified;		
of the Thermal	design features	description of the	many NRC-certified		General acceptance		
Design		thermal features	package designs		criteria not impacted		
			(NRC, 2013); detailed		by specifics of		
			description of thermal		technologies		
			features is expected to				
			be provided by the				
			applicant				
	Codes and	Codes and	Codes and standards	None expected	None identified;		
	standards	the thermal design	used to design the		General acceptance		
		and evoluation of	to be oveilable		by aposition of		
		the nackade			technologies		
	Content heat	Maximum decay	Detailed design	None expected	None identified:		
	load	heat load.	information is expected		General acceptance		
	specification	Methods and	to be provided by the		criteria not impacted		
	opoolinoution	codes used to	applicant		by specifics of		
		determine content			technologies		
		decay heat loads			3		
	Summary	Maximum,	Detailed design	None expected	None identified;		
	tables of	minimum, and	information is expected		General acceptance		
	temperatures	allowable	to be provided by the		criteria not impacted		
		temperatures of	applicant		by specifics of		
		package			technologies		
		components for					
		normal and					
		accident					
		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	Key information	Information	Potential	Potential guidance		
(NUREG-2216	ة, Chapter 3)	to be reviewed	availability	information needs	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		Key information	Information	Potential	Potential guidance		
(NUREG-2216	5, Chapter 3)	to be reviewed	availability	information needs	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 Packages for spent metal fuel						
Areas of review		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 Surface Temperati	Accessible ures	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 Key information Informati				Potential	Potential guidance		
(NUREG-2216	6, Chapter 3)	to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance		
(NUREG-2216	6, Chapter 3)	to be reviewed	availability	information needs	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							
pac	packages for spent metal fuel						
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	, Chapter 4)	to be reviewed	availability	information needs	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 Special	Codes and standards used for the containment design of the package Not applicable to spent metal fuel	Codes and standards used to design the package are expected to be available Not applicable	None expected Not applicable	None identified; General acceptance criteria not impacted by specifics of technologies Not applicable		
4.4.2 General Considerations	for damaged spent nuclear fuel Type AF fissile packages	Not applicable to spent metal fuel	Not applicable	Not applicable	Not applicable		

Table III-3. Info	Table III-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation						
pac	packages for spent metal fuel						
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	6, Chapter 4)	to be reviewed	availability	information needs	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	Туре В	Releasable source	Information available	Release rate	None identified. The		
Containment	transportation	term, maximum	on many NRC-	calculations and	review method calls for		
Evaluation under	packages	permissible	certified package	criteria used to	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	Key information	Information	Potential	Potential guidance		
(NUREG-2216	, Chapter 4)	to be reviewed	availability	information needs	gaps		
Normal		release rate,	designs (NRC, 2021,	verify cladding	calculates the maximum		
Conditions of		maximum	2013); Detailed	welds and fuel pins	permissible release rate		
Transport		permissible	containment	integrity to ensure	and maximum		
		leakage rate, and	evaluation is expected	sufficient	permissible leakage rate		
		conversion to the	to be provided by the	containment by the	in accordance with		
		reference air	applicant	transportation	ANSI N14.5. The		
		leakage rate		package of the	existing review method		
		calculated for		sodium-containing	is sufficient to evaluate		
		normal conditions		fuel under normal	metal fuel integrity		
		of transport in		conditions of	under normal conditions		
		accordance with		transport are to be	of transport.		
		ANSI N14.5		evaluated			
	0	(ANSI, 2014)		Dalaasawata	Niewe States 450 and The		
	Spent nuclear	Releasable source	Information available	Release rate	None identified. The		
	transportation	normissible	on many NRC-	calculations and	review method cans lor		
	nansportation			vorify clodding	ensuring the applicant		
	packages	maximum	2013): Detailed	welds and fuel nins	permissible release rate		
		nermissihle	containment	integrity to ensure	and maximum		
		leakage rate and	evaluation is expected	sufficient	permissible leakage rate		
		conversion to the	to be provided by the	containment by the	in accordance with		
		reference air	applicant	transportation	ANSI N14.5. The		
		leakage rate		package of the	existing review method		
		calculated for		sodium-containing	is sufficient to evaluate		
		normal conditions		fuel under normal	metal fuel integrity		
		of transport in		conditions of	under normal conditions		
		accordance with		transport are to be	of transport.		
		ANSI N14.5		evaluated	·		

Table III-3. Information to be reviewed and potential gaps for evaluating containment performance of transportation							
packages for spent metal fuel							
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	, Chapter 4)	to be reviewed	availability	information needs	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						
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	, Chapter 4)	to be reviewed	availability	information needs	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. MI 13309A031 Washington DC: U.S. Nuclear Regulatory Commission. 2013						

Table III-4. In	Table III-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation						
packages for spent metal fuel							
Areas o	f review	Key information	Information	Potential	Potential guidance		
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	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 o	f review	Key information	Information	Potential	Potential guidance	
	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. In	Table III-4. Information to be reviewed and potential gaps for evaluating shielding performance of transportation					
packages for spent metal fuel						
Areas o	of review	Key information	Information	Potential	Potential guidance	
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	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. "Stand < <u>https://compass.ast</u> NRC. "Certificate of Washington DC: U	ard Specification for U m.org/EDIT/html_anno Compliance for Radio	ranium Hexafluoride Enric <u>st.cgi?C996+20</u> > (Accesse active Materials Packages Commission 2021	ched to Less Than 5 percent ²³⁵ ed May 27, 2021). 2021. , Certificate No. 9225 for the N	U" AC-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	Key information	Information	Potential	Potential guidance		
(NUREG-2216	, Chapter 6)	to be reviewed	availability	information needs	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 <i>k</i> 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 mater	ial 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 (NUREG-2216	review 6, 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 <i>k</i> 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 (NUREG-2216	review 6, 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	Key information	Information	Potential	Potential guidance	
(NUREG-2216	, Chapter 6)	to be reviewed	availability	information needs	gaps	
			to be available at the			
			time of an application.			
	Evaluation of	Criticality	Information available on	None expected	None identified;	
	package	evaluation for an	many NRC-certified		General acceptance	
	arrays under	array of 2N	package designs (NRC,		criteria not impacted by	
	hypothetical	packages that is	2013). Although no prior		specifics of	
	accident	subcritical under	experience for spent		technologies	
	conditions	hypothetical	metal fuel package,			
		accident	information is expected			
		conditions	to be available at the			
			time of an application.			
	Package	Appropriate N	Information available on	None expected	None identified;	
	arrays results	value is used to	many NRC-certified		General acceptance	
	and criticality	ascertain the CSI	package designs (NRC,		criteria not impacted by	
	safety index		2013). Although no prior		specifics of	
	•		experience for spent		technologies	
			metal fuel package,			
			information is expected			
			to be available at the			
			time of an application.			

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

ble III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation						
packages for spent metal fuel						
Key information	Information	Potential	Potential guidance			
to be reviewed	availability	information needs	gaps			
Models and	Information is expected	None expected	None identified;			
	time of an application		General acceptance			
the korr calculations			specifics of			
representative of			technologies			
the physics in the			loonnologioo			
package						
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			
			60 GWd/MTU			
	reviewed and potentia t metal fuel Key information to be reviewed Models and analysis assumptions for the <i>k</i> _{eff} calculations representative of the physics in the package Validation of the depletion codes; Bias and bias uncertainty of the codes	reviewed and potential gaps for evaluating critt treat fuel Information availability Key information to be reviewed Information availability Models and analysis assumptions for the keff calculations representative of the physics in the package Information is expected to be available at the time of an application 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)	reviewed and potential gaps for evaluating criticality performance of treat fuel Key information to be reviewed Information availability Potential information needs Models and analysis Information is expected to be available at the time of an application None expected Models and analysis Information is expected to be available at the time of an application None expected Validation of the depletion 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 burnup limits are to be evaluated, given the potential lack of code validation data			

Table III-5. Info	Table III-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation						
packages for spent metal fuel							
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	5, Chapter 6)	to be reviewed	availability	information needs	gaps		
	Code validation— <i>keff</i> 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		
	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 re	-5. Information to be reviewed and potential gaps for evaluating criticality performance of transportation						
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packages for spent	metal fuel						
Areas of review	Key information	Information	Potential	Potential guidance			
(NUREG-2216, Chapter 6)	to be reviewed	availability	information needs	gaps			
ANS. American National Standards Institute	e/American Nuclear Societ	ty (ANSI/ANS) 8.1-1998 (R2007).	"Nuclear Criticality Safety	in Operations with			
Fissionable Materials Outside Reactors." La	a Grange Park, Illinois: Ar	nerican Nuclear Society. 2007.					
Jarrell, J. "A Proposed Path Forward for Tra	ansportation of High-Assa	y Low-Enriched Uranium." INL/EX	KT-18-51518. Idaho Falls, I	daho: Idaho National			
Laboratory. 2018.							
NRC. NUREG-0383, "Directory of Certifica	tes of Compliance for Rad	lioactive Materials Packages, Cer	tificates of Compliance." Vo	olume 2, Revision 28.			
ML13309A031. Washington, DC: U.S. Nuc	lear Regulatory Commissi	ion. 2013.					
NUREG/CR-6698, "Guide for Valida	ation of Nuclear Criticality	Safety Calculational Methodology	." Oak Ridge, Tennessee:	Science Applications			
International Corporation. U.S. Nuclear Reg	gulatory Commission. 200	01.					
NUREG/CR-5661, "Recommendation	ons for Preparing the Critic	cality Safety Evaluation of Transpo	ortation Packages." ORNL/	TM-11936. Oak Ridge, TN:			
Oak Ridge National Laboratory. U.S. Nucle	ar Regulatory Commission	n. 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	Key information	Information	Potential	Potential guidance		
(NUREG-22 ²	16, Chapter 7)	to be reviewed	availability	information needs	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						
p	ackages for spen	t metal fuel	I	Γ	1	
Areas	of review	Key information	Information	Potential	Potential guidance	
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance		
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	gaps		
7.4.4	Tensile	Acceptability of	Mechanical properties	None expected	None identified; General		
Mechanical	properties	material tensile	for commonly used		acceptance criteria not		
Properties		properties	packaging materials		impacted by specifics of		
			are available		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		
					the SPD calls for		
					examination of the		
					adequacy of information		
					in the application related		
					to mechanical properties		
					of those alternative		
					materials		
	Fracture	Acceptability of	Mechanical properties	None expected	None identified: General		
	resistance	material fracture	for commonly used	I	acceptance criteria not		
		toughness	packaging materials		impacted by specifics of		
		Ŭ	are available		technologies		
	Tensile	Acceptability of	Mechanical properties	None expected	None identified; General		
	properties and	the tensile	for commonly used		acceptance criteria not		
	creep of	properties and	aluminum alloys are		impacted by specifics of		
	aluminum alloys	creep of aluminum	available		technologies		
	at elevated	alloys			_		
	temperatures						

Table III-6. Ir	Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation							
р	packages for spent metal fuel							
Areas	of review	Key information	Information	Potential	Potential guidance			
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	gaps			
	Impact limiters	Acceptability of	Mechanical properties	None expected	None identified; General			
		the mechanical	for commonly used		acceptance criteria not			
		properties of the	impact limiter materials		impacted by specifics of			
		impact	are available		technologies			
		limiter materials						
7.4.5 Thermal P	roperties of	Thermal	Thermal properties for	Data characterizing	None identified. The			
Materials		properties of	commonly used	phases, phase	review method calls for			
		package	packaging materials	diagrams, heat	verification of the thermal			
		materials; Effect of	are available; Some	capacity, and	properties and the			
		degradation and	thermal properties of	thermal properties	change in these			
		anisotropic	fuel pin components,	of metal fuel and	properties from material			
		dependencies of	structural components,	converted waste	degradation. The existing			
		thermal properties	and metal fuel are	forms are limited	review method is			
			available (Leibowitz et		sufficient to deal with the			
			al., 1976; Janney,		availability of information			
			2018a, b; Janney and		related to metal fuel			
			Hayes, 2018; Janney		properties important to			
			et al., 2020, 2019)		the thermal analysis.			
7.4.6	Neutron-	Compositions and	Detailed evaluation of	None expected	None identified; General			
Radiation	shielding	geometries of	packaging materials is		acceptance criteria not			
Shielding	materials	shielding	expected to be		impacted by specifics of			
		materials;	provided by the		technologies			
		Acceptance	applicant					
		testing; Effect of						
		degradation and						
		temperature						
		dependencies of						
		shield						
		performance						

Areas of review (NUREG-2216, Chapter 7)Key information to be reviewedInformation availabilityPotential information needsPotential guidance gapsGamma- shielding materialsCompositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shieldDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical properties, fabricationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected acceptanceNone identified; General acceptance7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical properties, fabricationDetailed evaluation of provided by the applicantNone expected provided by the applicant7.4.7 Criticality ControlNeutron- absorbing (poison)Chemical properties, fabricationDetailed evaluation of provected to be provided by the applicantNone expected provided by the acceptance criteria not impacted by specifics of technologies	Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation								
Areas of reviewReymonizationInformationProtential guidance(NUREG-2216, Chapter 7)to be reviewedavailabilityinformation needsgapsGamma- shielding materialsCompositions and geometries of shielding materials;Detailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical omposition, physical and mechanical provided by the applicantDetailed evaluation of packaging materials is expected to be applicantNone expectedNone expected7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationDetailed evaluation of properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicant7.4.7 Criticality (uniticality)Neutron- composition, process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected home expected<		Areas of review Key information Information Detential Detential auidance							
(NOREG-2216, Chapter 7)To be reviewed to be reviewedavailabilityInformation needs availabilityggpsGamma- shielding materialsCompositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performanceDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical properties, fabricationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies		Di review	te be reviewed	mormation	Potential				
Campositions and shielding materialsCompositions and geometries of shielding materials; Acceptance testing; Effect of degradation and temperature dependencies of shield performanceDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies	(NUREG-22	Commo	Compositions and	Detailed evaluation of	None expected	yaps			
T.4.7 Criticality ControlNeutron- absorbing (poison) material specificationDetailed evaluation of packaging materials is expected to be applicantNone expected impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and provided by the applicantDetailed evaluation of packaging materials is expected to be packaging materials is expected to be packaging materials is expected to be packaging materials is expected to be provided by the applicantNone expected packaging materials is expected to be provided by the applicant7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected technologiesNone identified; General acceptance criteria not impacted by specifics of technologies		Gamma-	compositions and	Detailed evaluation of	None expected	None Identified, General			
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and mechanical provided by the applicantNone expected acceptance testing; Effect of degradation and temperature dependencies of shield performanceNone expected acceptanceNone identified; General acceptance technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical position, physical and mechanical provided by the applicantDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected packaging materials is expected to be provided by the applicant7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical provided by the applicantNone expected packaging materials is expected to be provided by the applicantNone expected packaging materials is expected to be provided by the applicant7.4.7 Criticality (poison) material specificationChemical provers, and minimum poison content of absorber materials; QualificationDetailed evaluation of provided by the applicant		matoriala	shielding	ovposted to be		impacted by specifics of			
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected acceptanceNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected technologiesNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected technologiesNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality (poison)Neutron- absorbing (poison)Composition, properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of process, and minimum poison content of absorber materials; Qualification		materials	materials.	provided by the		technologies			
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies			Accentance	applicant		lecinologies			
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and mechanical provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical provided by the applicantNone expected provided by the applicantNone expected provided by the applicant			testing: Effect of	applicant					
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and mechanical provided by the applicantDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical physical and properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected acceptance criteria not impacted by specifics of technologies			degradation and						
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies			temperature						
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and mechanical properties, fabricationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected materials is expected to be provided by the applicant			dependencies of						
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and mechanical provided by the applicantNone expected packaging materials is expected to be provided by the applicantNone expected materials is echnologiesNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expected technologiesNone identified; General acceptance criteria not impacted by specifics of technologies			shield						
7.4.7 Criticality ControlNeutron- absorbing (poison) material specificationChemical composition, physical and properties, fabricationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality ControlNeutron- absorbing (poison)Chemical composition, physical and properties, fabricationDetailed evaluation of packaging materials is expected to be provided by the applicantNone expectedNone identified; General acceptance criteria not impacted by specifics of technologies7.4.7 Criticality (poison)physical and mechanical properties, fabrication process, and minimum poison content of absorber materials; QualificationDetailed evaluation of packaging materials is expected to be provided by the applicant			performance						
Control absorbing (poison) physical and properties, fabrication process, and minimum poison content of absorber materials; Qualification	7.4.7 Criticality	Neutron-	Chemical	Detailed evaluation of	None expected	None identified; General			
(poison) material specificationphysical and mechanical properties, fabricationexpected to be provided by the applicantimpacted by specifics of technologiesimpacted by specificationproperties, fabrication process, and minimum poison content of absorber materials; Qualificationexpected to be provided by the applicantimpacted by specifics of technologies	Control	absorbing	composition,	packaging materials is		acceptance criteria not			
material specificationmechanical properties, fabrication process, and minimum poison content of absorber materials; Qualificationprovided by the applicanttechnologiesuiii <td< td=""><td></td><td>(poison)</td><td>physical and</td><td>expected to be</td><td></td><td>impacted by specifics of</td></td<>		(poison)	physical and	expected to be		impacted by specifics of			
specification properties, applicant fabrication process, and minimum poison content of absorber materials; Qualification		material	mechanical	provided by the		technologies			
fabrication process, and process, and minimum poison content of absorber materials; Qualification		specification	properties,	applicant					
process, and minimum poison content of absorber materials; Qualification			fabrication						
minimum poison content of absorber materials; Qualification			process, and						
content of absorber materials; Qualification			minimum poison						
absorber materials; Qualification			content of						
Qualification			absorber						
Qualification			materials;						
			Qualification						
			testing						
Computation of Level of credit Detailed evaluation of None expected None identified; General		Computation of	Level of credit	Detailed evaluation of	None expected	None Identified; General			
percent credit allowed for packaging materials is acceptance criteria not		for boron bosod	allowed for	packaging materials is		acceptance criteria not			
In boron-based absorber materials expected to be impacted by specifics of		nor poron-based	absorber materials	expected to be		technologies			
neutron provided by the technologies		neutron		provided by the		technologies			

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation							
packages for spent metal fuel							
Areas of	of review	Key information	Information	Potential	Potential guidance		
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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	Key information	Information	Potential	Potential guidance	
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	gaps	
7.4.9	Review	Coating	Detailed design	None expected	None identified; General	
Protective	guidance	specifications	information is expected		acceptance criteria not	
Coatings			to be provided by the		impacted by specifics of	
			applicant		technologies	
	Scope of	Purpose of the	Detailed design	None expected	None identified; General	
	coating	coating, lists the	information is expected		acceptance criteria not	
	application	components to	to be provided by the		impacted by specifics of	
		be coated, and the	applicant		technologies	
		expected				
		environmental				
	O a atim m	Conditions	Detailed and heating of	Non a como de d	Negeridentified, Osmanal	
	Coating	Coating	Detailed evaluation of	None expected	None identified; General	
	selection	tupo of primoro	packaging materials is		imported by aposition of	
		and toncost	expected to be		tochnologios	
		and topcoat,	provided by the		lechnologies	
		and ability of the	applicant			
		withstand the				
		conditions				
	Coating	Qualification	Detailed evaluation of	None expected	None identified: General	
	qualification	testing for coating	packaging materials is		acceptance criteria not	
	testing	performance in	expected to be		impacted by specifics of	
		accordance with	provided by the		technologies. The SRP	
		several standard	applicant		calls for evaluating any	
		ASTM (and			gualification testing for	
		possibly other)			the demonstration of	
		tests			coating performance.	

Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation								
р	packages for spent metal fuel							
Areas	of review	Key information	Information	Potential	Potential guidance			
(NUREG-22 ⁻	16, Chapter 7)	to be reviewed	availability	information needs	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 spen	packages for spent metal fuel							
Areas of review	Key information	Information	Potential	Potential guidance				
(NUREG-2216, Chapter 7)	to be reviewed	availability	information needs	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				

Table III-6.	Table III-6. Information to be reviewed and potential gaps for evaluating materials performance of transportation						
Areas (of review 16 Chapter 7)	Key information	Information availability	Potential	Potential guidance		
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 (Ui Cladding	nirradiated) Fuel	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		Key information	Information	Potential	Potential guidance		
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	gaps		
	Uncannistered	Cladding alloys	Stainless steel	Performance of	Additional guidance may		
	spent fuel	and maximum fuel	cladding performance	stainless steel	need to be developed to		
		burnup; Cladding	in transportation	cladding and spent	address mechanical		
		mechanical	packages may be	metal fuel under	properties of stainless		
		properties;	challenged by	transportation	steel and advanced		
		Effective cladding	sensitization,	environments is to	cladding materials for		
		thickness;	intergranular attack,	be evaluated;	metal fuel. The current		
		Maximum	stress corrosion	Advanced cladding	review method is limited		
		cladding	cracking, thermal	material properties	to guidance for the		
		temperature;	aging, and radiation	that can be used to	performance of zirconium		
		I nermal cycling	embrittlement	achieve high	alloy, aluminum alloy, or		
		during loading	(Alexander and	burnup are to be			
		operations;	Nansland, 1995;	evaluated,	lueis.		
		the server gee:	Cuantra et al., 2012;	especially material			
		Ligh burnun fuol	Sport motol fuel may	under the influence			
		monitoring and	ovporionco	of irradiation			
		accessment:	degradation during				
		Release fractions	transportation				
			particularly oxidation				
			hydriding				
			fragmentation and				
			restructuring-swelling				
			(Guenther et al., 1996).				
	Cannistered	Performance of	Detailed evaluation of	None expected	None identified; General		
	spent fuel	the fuel can for	the fuel can		acceptance criteria not		
		damaged fuel	performance is		impacted by specifics of		
		Ŭ	expected to be		technologies		
			provided by the		_		
			applicant				

Table III-6. II	6. Information to be reviewed and potential gaps for evaluating materials performance of transportation packages for spent metal fuel					
Areas	of review	Key information	Information	Potential	Potential guidance	
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	gaps	
7.4.15 Bolting N	laterial	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	Key information	Information	Potential	Potential guidance
(NUREG-2216, Chapter 7)	to be reviewed	availability	information needs	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), Rareearth 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			1		
Areas of review		Key information	Information	Potential	Potential guidance	
(NUREG-2216	5, Chapter 2)	to be reviewed	availability	information needs	gaps	
2.4.1 Description	General	Drawings and	Information available on	None expected	None identified;	
of Structural		descriptive	many NRC-certified		General acceptance	
Design		information	package designs,		criteria not impacted	
		including weights	particularly the TN-FSV		by specifics of	
		and centers of	and NAC-LWT		technologies	
		gravity	packages for transport			
			of TRISO fuel			
			(NRC, 2021, 2014,			
			2013); Detailed			
			description of structural			
			design is expected to be			
			provided by the			
	lala a tifi a ati a a af	Cadaa and			Non a identificati	
	Identification of	Codes and	Codes and standards	None expected	None Identified;	
	coues and	standards used for	used to design the		General acceptance	
	standards for	the package	to be evoluble		chiena not impacted	
	package		to be avaliable			
2.4.2 Conorol	Minimum		Specific peckage	None expected	None identified:	
Z.4.2 General Dequirements for				None expected	Conoral accontance	
All Dackagos	package size	of the nackage	ovported to be provided		critoria not impacted	
All Fackages		must not be less	by the applicant		by specifics of	
		than $10 \text{ cm} [4 \text{ in}]$	by the applicant		technologies	
		The nackade	Detailed nackade	None expected	None identified:	
	indicating	closure system	design information is		General accentance	
	feature	must incornorate a	expected to be provided		criteria not impacted	
		tamper-indicating	by the applicant		by specifics of	
		feature			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 Key information Information Potential Potential guid						
(NUREG-2216	6, Chapter 2)	to be reviewed	availability	information needs	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. Info	able III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages					
for	spent TRISO fuel					
Areas of review		Key information	Information	Potential	Potential guidance	
(NUREG-2216	, Chapter 2)	to be reviewed	availability	information needs	gaps	
2.4.5 Normal	Heat	Maximum	Structural performance	None expected	None identified;	
Conditions of		temperature,	of the package under		General acceptance	
Transport		maximum	the heat-loading		criteria not impacted	
		pressure, and	condition is expected to		by specifics of	
		thermal stress	be provided by the		technologies	
		under the heat-	applicant			
		loading condition				
	Cold	Maximum	Structural performance	None expected	None identified;	
		temperature,	of the package under		General acceptance	
		minimum	the cold condition is		criteria not impacted	
		internal pressure,	expected to be provided		by specifics of	
		and residual stress	by the applicant		technologies	
		under the				
		cold condition				
	Reduced	Effects of reduced	Effects of reduced	None expected	None identified;	
	external	external pressure	external pressure is		General acceptance	
	pressure	on the internal and	expected to be		criteria not impacted	
		external	described and		by specifics of	
		pressures of the	evaluated by the		technologies	
		package and the	applicant			
		containment				
		system				
	Increased	Effects of	Effects of increased	None expected	None identified;	
	external	increased	external pressure is		General acceptance	
	pressure	external pressure	expected to be		criteria not impacted	
		on the internal and	described and		by specifics of	
		external	evaluated by the		technologies	
		pressures of the	applicant			
		package and the				
		containment				
		system				

Table III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for sport TPISO fuel For sport TPISO fuel						
Areas of review Key information Information Potential Potential quidance						
(NURFG-2216, Chapter 2)		to be reviewed	availability	information needs	ans	
	Vibration and fatigue	Effects of vibration normally incident to transport; Fatigue under the combined stresses from vibration,	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	temperature, and pressure loads 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	
	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. Info for	Table III-7. Information to be reviewed and potential gaps for evaluating structural integrity of transportation packages for spent TRISO fuel					
Areas of review Key information Information				Potential	Potential guidance	
(NUREG-2216	5, Chapter 2)	to be reviewed	availability	information needs	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. MI 13309A031 Washington, DC: U.S. Nuclear Regulatory Commission 2013					

Table III-8. Info	able III-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation					
рас	packages for spent TRISO fuel					
Areas of review		Key information	Information	Potential	Potential guidance	
(NUREG-2216	5, Chapter 3)	to be reviewed	availability	information needs	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	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					
Areas of (NUREG-2216	review 6, 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							
рас	packages for spent TRISO fuel						
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	<u>5, Chapter 3)</u>	to be reviewed	availability	information needs	gaps		
3.4.3 General	Evaluation by	Elements of the	Specific analyses used	None expected	None identified;		
Considerations	analyses	analysis used for	to evaluate the		General acceptance		
for Thermal		thermal evaluation	package are expected		criteria not impacted		
Evaluations			to be provided by the		by specifics of		
			applicant		technologies		
	Evaluation by	Elements of the	Specific tests used to	None expected	None identified;		
	Tests	test used for	evaluate the package		General acceptance		
		thermal evaluation	are expected to be		criteria not impacted		
			provided by the		by specifics of		
			applicant		technologies		
	Confirmatory	Rigor of the	Detailed thermal	None expected	None identified;		
	analyses	confirmatory	evaluation is expected		General acceptance		
		analysis	to be provided by the		criteria not impacted		
			applicant		by specifics of		
					technologies		
	Effects of	Uncertainties in	Detailed thermal	None expected	None identified;		
	uncertainties	thermal	evaluation is expected		General acceptance		
		and structural	to be provided by the		criteria not impacted		
		properties of	applicant		by specifics of		
		materials, test			technologies		
		conditions and					
		diagnostics, and					
		analytical methods					
	Conservatisms	Conservatisms	Detailed thermal	None expected	None identified;		
		associated with	evaluation is expected		General acceptance		
			to be provided by the		criteria not impacted		
		models and their	applicant		by specifics of		
		effects on the			technologies		
		satety parameters					

Table III-8. Information to be reviewed and potential gaps for evaluating thermal performance of transportation						
packages for spent TRISO fuel						
	Chapter 2)	to be reviewed	Information	Potential	Potential guidance	
2.4.4 Evoluction of	, Chapter 3)	Thormol model	Detailed thermal	More expected	gaps None identified:	
Surface Temperat	Accessible			None expected	Conorol accontance	
	lies	used ioi	to be provided by the		oritoria not imposted	
					by appointing of	
		surface	applicant		technologies	
		temperatures			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	Detailed thermal evaluation is expected to be provided by the applicant	None expected	None identified; General acceptance criteria not impacted by specifics of technologies	
		components under the cold condition				
	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								
pac	packages for spent TRISO fuel							
Areas of	review	Key information	Information	Potential	Potential guidance			
(NUREG-2216	5, Chapter 3)	to be reviewed	availability	information needs	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 C Washington, DC: U.S. "Certificate of (Washington, DC: U.S. NUREG–0383	ompliance for Radioad Nuclear Regulatory C Compliance for Radioa Nuclear Regulatory C , "Directory of Certifica	ctive Materials Packages, C Commission. 2021. active Materials Packages, Commission. 2014. ates of Compliance for Rac	Certificate No. 9225 for the NAC Certificate No. 9253 for the TN lioactive Materials Packages, C	C-LWT Package." Revision 71 I-FSV Package." Revision 13. certificates of Compliance." Vo	. ML21078A200. ML14167A316. blume 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						
pac	kages for spent	TRISO fuel				
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	, Chapter 4)	to be reviewed	availability	information needs	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	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	Type AF fissile packages	Not applicable to spent TRISO fuel	Not applicable	Not applicable	Not applicable	
for Containment Evaluations	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							
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	Key information	Information	Potential	Potential guidance	
(NUREG-2216	5, Chapter 4)	to be reviewed	availability	information needs	gaps	
4.4.4	Туре В	Releasable source	Existing NRC-certified	None expected	None identified;	
Containment	transportation	term, maximum	TN-FSV and NAC-LWT		General acceptance	
Evaluation	packages	permissible	packages (NRC, 2021,		criteria not impacted	
Under		release rate,	2014) provide		by specifics of	
Hypothetical		maximum	applicable containment		technologies	
Accident		permissible	evaluation under			
Conditions		leakage rate, and	hypothetical accident			
		conversion to the	conditions;			
		reference air	Detailed package			
		leakage rate	containment evaluation			
		nypoinelical	accident conditions is			
			by the applicant			
		accordance with	by the applicant			
	Spent nuclear	Releasable source	Existing NBC-certified	None expected	None identified:	
	fuel	term maximum	TN-FSV and NAC-I WT		General acceptance	
	transportation	permissible	packages (NRC, 2021.		criteria not impacted	
	packages	release rate.	2014) provide		by specifics of	
	1 0	maximum	applicable containment		technologies	
		permissible	evaluation under		Ŭ	
		leakage rate, and	hypothetical accident			
		conversion to the	conditions;			
		reference air	Detailed package			
		leakage rate	containment evaluation			
		calculated for	under hypothetical			
		hypothetical	accident conditions is			
		accident	expected to be provided			
		conditions in	by the applicant			
		accordance with				
		ANSI N14.5				

Table III-9. Information to be reviewed and potential gaps for evaluating containment performance of transportation								
packages for spent TRISO fuel								
Areas of	review	Key information	Information	Potential	Potential guidance			
(NUREG-2216	6, Chapter 4)	to be reviewed	availability	information needs	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 N14.5-20	14, "American Nation	al Standard for Radioactiv	e Materials–Leakage Tests on P	ackages for Shipment." New	York, New York: American			
NRC "Certificate of C	Stitute. 2014.	active Materials Packages	Certificate No. 9225 for the NAC	-I WT Package " Revision 71	MI 210784200			
Washington, DC: U.S	Nuclear Regulatory	Commission, 2021.	Certificate No. 9223 for the NAC	-LIVI I ackage. Revision / I	. WE21070A200.			
"Certificate of	Compliance for Padic	active Materials Package	Cortificato No. 0253 for the TN	ESV Package " Povision 13	MI 1/167A316			

. "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						
ра	ckages for spent	TRISO fuel		1		
Areas o	f review	Key information	Information	Potential	Potential guidance	
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	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		Key information	Information	Potential	Potential guidance			
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	gaps			
	Gamma sources	Gamma source	Detailed shielding	None expected	None identified;			
		strengths and	evaluation is expected		General acceptance			
		spectra for the	to be provided by the		criteria not impacted			
		package contents	applicant		by specifics of			
					technologies			
	Neutron sources	Neutron source	Detailed shielding	None expected	None identified;			
		strengths and	evaluation is expected		General acceptance			
		spectra for the	to be provided by the		criteria not impacted			
		package contents	applicant		by specifics of			
					technologies			
5.4.3 Shielding	Configuration of	Dimensions and	Information available on	None expected	None identified;			
Model and	source and	materials	many NRC-certified		General acceptance			
Model	shielding	properties of the	package designs (NRC,		criteria not impacted			
Specifications		package contents,	2021, 2014, 2013);		by specifics of			
		radioactive	Detailed shielding		technologies			
		sources in the	evaluation is expected					
		contents, and the	to be provided by the					
		packaging	applicant					
		components						
	Material	Material properties	Detailed design	None expected	None identified;			
	properties	(e.g., composition,	Information is expected		General acceptance			
		mass densities,	to be provided by the		criteria not impacted			
		and atom	applicant		by specifics of			
		densities) of			technologies			
		packaging						
		components,						
		package contents,						
		and the						
		conveyance						

Table III-10. Inf	Table III-10. Information to be reviewed and potential gaps for evaluating shielding performance of transportation							
ра	packages for spent TRISO fuel							
Areas of review		Key information	Information	Potential	Potential guidance			
(NUREG-221	6, Chapter 5)	to be reviewed	availability	information needs	gaps			
5.4.4 Shielding	Methods	Methods used for	Detailed shielding	None expected	None identified;			
Evaluation		the shielding	evaluation is expected		General acceptance			
		evaluations under	to be provided by the		criteria not impacted			
		normal and	applicant		by specifics of			
		accident			technologies			
			Detailed abiat divers		Niewe isterational			
	Code input and	Key input data and		None expected	None identified;			
	oulpul data	shielding	to be provided by the		General acceptance			
		evaluations	applicant		by specifics of			
		CValuations	applicant		technologies			
	Fluence-rate-to-	Accuracy and	Detailed shielding	None expected	None identified:			
	radiation-level	acceptance of the	evaluation is expected		General acceptance			
	conversion	conversion factors	to be provided by the		criteria not impacted			
	factors		applicant		by specifics of			
					technologies			
	External	External radiation	Detailed shielding	None expected	None identified;			
	radiation levels	levels under	evaluation is expected		General acceptance			
		normal and	to be provided by the		criteria not impacted			
		accident	applicant		by specifics of			
		conditions			technologies			
	Confirmatory	Rigor of the	Detailed shielding	None expected	None identified;			
	analyses	confirmatory	evaluation is expected		General acceptance			
		analyses	to be provided by the		criteria not impacted			
			applicant		by specifics of			
ASTM COOG "Stand	rd Specification for Lin		ad to Loop Then 5 percent 2351.		technologies			
<https: compass.astr<="" td=""><td>m org/EDIT/html appot</td><td>cdi2C996+20> (Accessed</td><td>May 27 2021) 2021</td><td></td><td></td></https:>	m org/EDIT/html appot	cdi2C996+20> (Accessed	May 27 2021) 2021					

(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								
pac	packages for spent TRISO fuel							
Areas of	review	Key information	Information	Potential	Potential guidance			
(NUREG-2216	5, Chapter 6)	to be reviewed	availability	information needs	gaps			
6.4.1 Description	Packaging	Design features	Information available on	None expected	None identified;			
of Criticality	design	important for	many NRC-certified		General acceptance			
Design	features	criticality safety	package designs,		criteria not impacted by			
			particularly the		specifics of			
			TN-FSV and NAC-LWT		technologies			
			packages for transport					
			of TRISO fuel					
			(NRC, 2021, 2014,					
			2013); Detailed					
			description of criticality					
			features is expected to					
			be provided by the					
			applicant					
	Codes and	Codes and	Codes and standards	None expected	None identified;			
	standards	standards used in	used to design the		General acceptance			
		all aspects of	package are expected		criteria not impacted by			
		the criticality	to be available		specifics of			
		design and			technologies			
		evaluation						
	Summary table	Maximum value of	Detailed design	None expected	None identified;			
	of criticality	<i>k</i> eff, uncertainty,	information is expected		General acceptance			
	evaluations	bias and bias	to be provided by the		criteria not impacted by			
		uncertainty for all	applicant		specifics of			
		relevant cases;			technologies			
		Number of						
		packages						
		evaluated in the						
		array cases						

Table III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation packages for spent TRISO fuel						
Areas of (NUREG-2216	review 6, 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 mater	rial 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							
Areas of	review	Key information	Information	Potential	Potential guidance		
(NUREG-2216	, Chapter 6)	to be reviewed	availability	information needs	gaps		
	Demonstration of maximum reactivity	Analyses demonstrate the maximum <i>k_{eff}</i>	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							
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Areas of (NUREG-2216	review 6, 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						
pac	kages for spent	TRISO fuel				
Areas of	review	Key information	Information	Potential	Potential guidance	
(NUREG-2216	, Chapter 6)	to be reviewed	availability	information needs	gaps	
Evaluations	and applicability	computer codes for criticality calculations against fitting	many NRC-certified package designs, particularly the TN-FSV and NAC-LWT	benchmark data and applicability of existing criticality codes and methods	review method calls for verifying the applicant has benchmarked the computer codes used	
		critical experiments	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)	for spent TRISO fuel with initial enrichments between 5 and 20 weight percent ²³⁵ U are to be evaluated	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		Key information	Information	Potential	Potential guidance			
(NUREG-2216	5, Chapter 6)	to be reviewed	availability	information needs	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								
pacl	packages for spent TRISO fuel							
Areas of I	review	Key information	Information	Potential	Potential guidance			
(NUREG-2216,	, Chapter 6)	to be reviewed	availability	information needs	gaps			
	Model assumptions	Models and analysis assumptions for the <i>k</i> _{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							
Areas of review Key information Information Potent					Potential guidance		
(NUREG-2216	6, Chapter 6)	to be reviewed	availability	information needs	gaps		
	Code validation— <i>keff</i> 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 III-11. Information to be reviewed and potential gaps for evaluating criticality performance of transportation									
packages for spent	packages for spent TRISO fuel								
Areas of review	Key information	Information	Potential	Potential guidance					
(NUREG-2216, Chapter 6)	to be reviewed	availability	information needs	gaps					
ANS. American National Standards Institute	e/American Nuclear Societ	y (ANSI/ANS) 8.1-1998 (R2007).	"Nuclear Criticality Safety	in Operations with					
Fissionable Materials Outside Reactors." La	a Grange Park, Illinois: An	nerican Nuclear Society. 2007.							
Jarrell, J. "A Proposed Path Forward for Tra	ansportation of High-Assay	Low-Enriched Uranium." INL/EX	XT-18-51518. Idaho Falls, I	Idaho: Idaho National					
Laboratory. 2018.									
NRC. "Certificate of Compliance for Radioa	ctive Materials Packages,	Certificate No. 9225 for the NAC	-LWT Package." Revision 7	71. ML21078A200.					
Washington, DC: U.S. Nuclear Regulatory	Commission. 2021.								
"Safety Evaluation Report for Model	No. Versa-Pac Package (Certificate of Compliance No. 934	12 Revision No. 15." ML20	139A037. Washington, DC:					
U.S. Nuclear Regulatory Commission. 2020).								
"Certificate of Compliance for Radio	active Materials Packages	, Certificate No. 9253 for the TN-	FSV Package." Revision 1	3. ML14167A316.					
Washington, DC: U.S. Nuclear Regulatory	Commission. 2014.								
NUREG–0383, "Directory of Certific	ates of Compliance for Ra	dioactive Materials Packages, Ce	ertificates of Compliance."	Volume 2, Revision 28.					
ML13309A031. Washington, DC: U.S. Nuc	lear Regulatory Commissi	on. 2013.							
NUREG/CR-6698, "Guide for Valida	. 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, "Recommendation	ons for Preparing the Critic	ality Safety Evaluation of Transp	ortation Packages." ORNL/	/TM-11936. Oak Ridge, TN:					
Oak Ridge National Laboratory. U.S. Nucle	ar Regulatory Commission	. 1997.							

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation							
p	packages for spent TRISO fuel						
Areas of review		Key information	Information	Potential	Potential guidance		
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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						
p	packages for spent TRISO fuel					
		Key information	Information	Potential	Potential guidance	
(NUREG-221	6, Chapter 7)	to be reviewed		Information needs	gaps	
			Specific code case	None expected	None Identified; General	
	use/acceptability	ASIVIE code cases	te he provided by the		acceptance criteria not	
			to be provided by the		Impacted by specifics of	
		Construction of	Applicant Detailed design and	None expected	Lechnologies	
	NON-ASIVIE		Detailed design and	None expected	None Identified; General	
	coue	non-ASIVIE coue	information is expected		imported by aposition of	
	components	components	to be provided by the		tochnologios	
			applicant		lecinologies	
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	

Table III-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation						
Areas o	Areas of review Key information Information Potential Potential quida					
(NUREG-221	6. Chapter 7)	to be reviewed	availability	information needs	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	of review	Key information	Information	Potential	Potential guidance	
(NUREG-22)	Fracture resistance	Acceptability of material fracture toughness	Mechanical properties for commonly used packaging materials are available	None expected	gaps 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	of review	Key information	Information	Potential	Potential guidance	
(NUREG-22	6, Chapter 7)	to be reviewed	availability	information needs	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 Packages for spent TRISO fuel						
Areas of (NUREG-221	of review 16, 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		Key information	Information	Potential	Potential guidance
(NUREG-2216, Chapter 7)		to be reviewed	availability	information needs	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 nackages for spent TRISO fuel					
Areas of review		Key information	Information	Potential	Potential guidance
(NUREG-221	6, Chapter 7)	to be reviewed	availability	information needs	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		Key information	Information	Potential	Potential guidance
(NUREG-2216, Chapter 7)		to be reviewed	availability	information needs	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.	12. Information to be reviewed and potential gaps for evaluating materials performance of transportation				
packages for spent TRISO fuel					
Areas of review		Key information	Information	Potential	Potential guidance
(NUREG-2216, Chapter 7)		to be reviewed	availability	information needs	gaps
7.4.14 Spent	Spent fuel	Classification of	Information is expected	None expected	None identified; General
Nuclear Fuel	classification	damaged,	to be available at the		acceptance criteria not
		undamaged, and	time of an application		impacted by specifics of
		intact fuel			technologies
	Uncannistered	Cladding alloys and	Although degradation	Performance of the	Additional guidance may
	spent fuel	maximum fuel	of TRISO fuel has not	coating layers on	need to be developed to
		burnup; Cladding	been reported	TRISO fuel under	address the mechanical
		mechanical	associated with	transportation	properties of coating
		properties;	transportation	environments is to	layers of TRISO fuel.
		Effective cladding	conditions, information	be evaluated	The current review
		thickness;	in the literature is		method is limited to
		Maximum cladding	limited		guidance for the
		temperature;			performance of
		I nermal cycling			zirconium alloy,
		during loading			aluminum alloy, or
		operations;			stainless steel-clad UO ₂
		Composition of the			Tuels, which is not
		cover gas; High			applicable to TRISO luel.
		burnup luel			It is necessary to
		nonitoring and			or equivalent cladding
		Release fractions			functions are required in
					TRISO fuel.
	Cannistered	Performance of the	Detailed evaluation of	None expected	None identified; General
	spent fuel	fuel can for	the fuel can		acceptance criteria not
		damaged fuel	performance is		impacted by specifics of
			expected to be		technologies
			provided by the		
			applicant		

Table III-12. II	II-12. Information to be reviewed and potential gaps for evaluating materials performance of transportation				
packages for spent TRISO fuel					
Areas of review		Key information	Information	Potential	Potential guidance
(NUREG-22	16, Chapter 7)	to be reviewed	availability	information needs	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."					

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