INFORMATION GAPS AND POTENTIAL INFORMATION NEEDS ASSOCIATED WITH TRANSPORTATION OF FRESH (UNIRRADIATED) ADVANCED REACTOR FUEL TYPES

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ABSTRACT

This report documents technical information availability and potential information gaps related to technical areas that the U.S. Nuclear Regulatory Commission (NRC) staff would be considered during reviews of transportation packages intended for use with fresh advanced reactor fuel (ARF) types, including nuclear metal fuel and tristructural isotropic (TRISO), as well as uranium hexafluoride (UF_6) with possible enrichment up to 20 weight percent U-235 containing recycled uranium. For each of the review topics outlined in NUREG–1609, publicly available literature was reviewed to establish potential information gaps for possible package contents. The Versa-Pac is certified by NRC to transport fresh TRISO fuel, which is a type of fuel that is well documented in the literature. Based on a review of the Versa-Pac safety analysis report and available literature, limited specific information gaps were identified for transportation package certification reviews for fresh TRISO fuel. Although some characteristics of nuclear metal fuel, including thermal and mechanical properties, are well documented, potential information gaps were identified as they relate to transportation of fresh metal fuel. Testing and analyses that document the performance of sodium-containing fresh metal fuel pins or assemblies during normal and accident conditions, needed to support structural evaluations, are limited. For materials used in constructing metal fuel, parameters necessary to fully assess fuel pin or assembly performance under accident conditions are limited, including phases, heat capacity, and mechanical and thermal properties. In addition, understanding of the effects on thermal expansion or bonding during extreme temperature conditions that could be experienced under accident conditions would be improved with further information. Additionally, for sodium-containing fuel assemblies, design information related to cladding welds, total strain absorption energy, yield stress, ultimate stress as a function of temperature, release rate calculation, and criteria used to verify fuel pin weld integrity by nondestructive methods to ensure containment of the sodium-containing fuel under normal conditions of transport is limited for the evaluation of the package performance against Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71 requirements and other containment requirements. UF_6 produced by reprocessing of irradiated uranium can contain concentrations of other uranium isotopes, transuranic nuclides, fission product impurities, and daughter products of these species. Standard cylinders listed for transportation of UF_6 are listed in American National Standards Institute (ANSI) N14.1, including the 30B cylinder, credited as a component for proposed packaging intended for the safe transport of up to 1,600 kg of UF_6 with enrichments of up to 20 weight percent U-235. For UF_6 transportation package certification, information related to structural, containment, and thermal evaluations are available and well documented. However,UF_6 used for fabricating metal and TRISO fuels could originate from reprocessed uranium, and as such its specific composition and quantity of radiological source terms (gamma and neutron), and other radiation sources, impurities, etc., would need full assessment in order to establish shielding requirements for transportation of larger quantities of reprocessed UF_6, metal fuel, and TRISO fuel with up to 20 weight percent U-235. For TRISO, metal fuel, and UF_6, criticality experiments for high-essay low-enriched uranium (HALEU) assays that consider representative transportation package configurations with contents to 20 weight percent U-235 are limited, and could result in an applicant's use of conservative design assumptions, or obtaining additional criticality data in order to reduce uncertainty with existing benchmark data used to validate criticality evaluations.

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ABBREVIATIONS/ACRONYMS

AGR	advanced gas reactor
ANSI	American National Standards Institute
ARF	advanced reactor fuel
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
10 CFR	10 of the *Code of Federal Regulations*
CNWRA®	Center for Nuclear Waste Regulatory Analyses®
CoC	Certificate of Compliance
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EBR-II	Experimental Breeder Reactor-II
EPRI	Electric Power Research Institute
FHRs	fluoride salt-cooled high-temperature reactors
FFTF	Fast Flux Test Facility
HAC	hypothetical accident conditions
HALEU	high-essay low-enriched uranium
HF	hydrogen fluoride
HTGR	high-temperature gas-cooled reactor
LWR	light water reactor
NCT	normal conditions of transport
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PSP	protective structural packaging
SAR	Safety Analysis Report
SiC	silicon carbide
SRP	Standard Review Plan
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 Background

To help prepare for regulatory interactions and potential applications for non-light water reactor (LWR) technologies, the U.S. Nuclear Regulatory Commission (NRC) staff seeks to assess the availability of technical information necessary to assist safety reviews of transportation packages that would be used for transportation of fresh advanced reactor fuels. NRC safety reviews for transportation package certification for advanced reactor fuel (ARF) types would span technical review topics outlined in NUREG–1609 and would utilize information contained in NRC technical guidance and regulations for transportation. Potential ARF types that could be subject to NRC regulation in the future include metal fuels, tristructural isotropic (TRISO) fuels for high-temperature gas-cooled reactors (HTGRs), and molten fuel salt, as well as precursor uranium hexafluoride (UF_6) with up to 20 weight percent U-235 used in fabrication of fuel. Early identification of potential information gaps for technical areas important to transportation package certification, based on available literature for similar non-LWR fuel, facilitates the development of information needs related to regulating transportation of fresh ARF.

1.2 Purpose and Scope

The Center for Nuclear Waste Regulatory Analyses (CNWRA®) has been tasked with reviewing publicly available literature, applicable NRC regulations and guidance, and previously identified technical challenges that address relevant safety review topics, to establish potential information gaps related to technical issues important to package certification for transportation of fresh (unirradiated) ARF types. The ARF types considered for this report are nuclear metal fuel and TRISO fuel, along with fuel fabrication precursor UF_6 with possible enrichment up to 20 weight percent U-235. The following NRC regulations and guidance were consulted:

- 10 CFR Part 71, Packaging and Transportation of Radioactive Material; and
- NUREG–1609, Standard Review Plan for Transportation Packages for Radioactive Material (NRC, 1999), which addresses safety topics such as criticality, shielding, structural, containment, and thermal functions of packages, and describes the safety requirements that must be satisfied in order for an applicant to obtain a Certificate of Compliance (CoC).

The scope of this report includes: (i) identifying information to be reviewed for the evaluation of packages for transporting fresh ARF types to demonstrate that the package provides adequate structural integrity, thermal, containment, shielding, and criticality protection under normal and accident conditions; (ii) conducting a review of literature to identify and assess available information related to key topics, and identifying information gaps within the context of the safety evaluation topics; and (iii) discussing potential research needs to address the information gaps identified under (i) and (ii). Section 2 of the report covers (i) and (ii) and Section 3 covers (iii).

2 REVIEW AND ASSESSMENT OF LITERATURE INFORMATION FOR SAFETY EVALUATION ASSOCIATED WITH TRANSPORTATION OF FRESH ARF TYPES

Advanced reactor fuel (ARF) types considered for this report are nuclear metal fuel and TRISO fuel, along with UF_6 as a fuel fabrication precursor, all with possible enrichments up to 20 weight percent U-235.

2.1 Fresh Metal Fuel

Hall et al. (2019a) described the typical configuration of the fresh metal fuel pin. Some features important for the safety functions, which are very different from the non-light water reactor (LWR) fuel rod, are highlighted in this section. The metal fuel components include metal fuel slugs, cladding, a thermal bond material between fuel and cladding, a gas plenum, and end plugs. The thermal bond material is usually sodium. During metal fuel pin fabrication, different bonding techniques are used to remove any gaps, voids, or defects present in the annulus region between the fuel slug and cladding (Burkes et al., 2009). Although there are no reported standards, such as American Society for Testing and Materials (ASTM), that define critical characteristics of bonding processes, the bonding process is relied upon to ensure adequate in-reactor performance, including heat transfer between the fuel and the cladding until the fuel swells to contact the cladding. In comparison, for the LWR, the fuel cladding gap is filled with helium gas to improve the conduction of heat from the fuel to the cladding without any thermal bond material such as sodium. As a result, different from LWR fuel, the cladding and the thermal bonding between fuel and cladding need to be jointly considered in structural evaluations. In addition to the typical fuel configuration described in Hall et al. (2019a), metal fuel design can vary such as the fast neutron reactor design proposed by Oklo Inc. (2020) that uses heat pipes to transfer heat from the reactor core to a supercritical carbon dioxide power conversion system to generate electricity. The U-10Zr metal fuel, which is the component with the most mass of the entire heat pipe, lies only at the bottom third of the structure. This uneven mass distribution along the length of the pipe makes the structure very nonsymmetric.

Fresh fuel made from uranium ore with no prior history of irradiation would presumably only need a package that meets requirements for shipping Type AF fissile material. However, metal fuel could be based on high-essay low-enriched uranium (HALEU), with enrichments ranging from 5 to 20 percent, and fabricated from electrochemically processed Experimental Breeder Reactor-II (EBR-II) spent fuel following a process described in Hall et al. (2019c). HALEU derived from an irradiated uranium source such as EBR-II spent fuel can contain Pu-239 and Pu-240, which have A2 values of 0.027 Ci (per 10 CFR Part 71, A2 means the maximum activity of radioactive material, other than special form material, low specific activity, and surface contaminated object material, permitted in a Type A package), and therefore transport of this fresh metal fuel could require a Type BF container to handle residual radioactivity (Eidelpes et al., 2019). The evaluation in this section includes both Types AF and BF packages. Previous reports (Hall et al., 2019a,b) reviewed available operating experience and identified potential challenges with transportation of fresh metal fuel. A review of literature for operating reactors did not identify certified packages that were already used for transportation of fresh metal fuel, nor possible scenarios defining exact package contents. This section provides more detailed information for potential safety reviews of transportation of fresh metal fuel within the context of the package safety evaluation topics of structural, thermal, containment, shielding, and criticality performance with focus on the fresh fuel.

2.1.1 Structural evaluation

The objective of the structural review is to evaluate the information presented in a potential application relevant to structural performance of the package design under normal conditions of transport (NCT) and hypothetical accident conditions (HAC). Information to be reviewed to evaluate structural performance includes the following:

1) Structural design

The structural design includes (i) description of the principal structural components of the transportation package that serves as the primary impact and thermal protection for the contents (e.g., outer shell, inner shell, outerpack, the impact limiters), the component that provides for lifting, stacking, and tie down during transportation, the fuel contents (i.e., fuel assembly), and the structure that protects and restrains the fuel contents during all transport conditions; (ii) criteria to demonstrate that the package maintains its structural, thermal, containment, shielding, and criticality functions through both NCT and HAC scenarios; (iii) weights (e.g., gross weight, tare weight), dimensions (e.g., outer dimensions, inner dimensions that accommodate the fuel contents), and centers of gravity; (iv) general standards; and (v) lifting and tie-down standards for all packages to demonstrate compliance with requirements of 10 CFR 71.45 including failure under excessive load. Testing, standard engineering calculations, and computer simulations are commonly used to evaluate structural design. The fabrication, assembly, testing, maintenance, and operation of package are accomplished with the use of generally accepted codes and standards, such as American Society of Mechanical Engineers (ASME), ASTM, and American Welding Society (AWS). ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in the mechanical design and stress analysis of the fuel pin/assembly to withstand the applied loads. Although structural design information is available from many NRC-certified packages (NRC, 2013), none is specifically for sodium-containing fresh metal fuel pins or assembly. It is uncertain if the generally accepted codes and standards are sufficient for structural design analysis of fresh metal fuel with nonsymmetrical contents such as the Oklo application (Oklo Inc., 2020) or other future designs. Information gap for fresh metal fuel package would depend on metal fuel design at the time of an application.

2) Tests and analysis under normal conditions of transport

Tests and analysis under NCT include (i) cold and heat at design temperatures between −40 °C [−40 °F] and 38 °C [100 °F] per 10 CFR 71.71(c)(1,2) to demonstrate no brittle fracture and no impact from differential thermal expansion; (ii) reduced external pressure; (iii) increased external pressure (iv) vibration to demonstrate no impact on structural performance; (v) water spray; (vi) free drop for 1 to 4 feet to demonstrate that the geometric form of the package contents would not be substantially altered; (vii) corner drop; (viii) compression from stacking; and (ix) penetration.

Table 2-1 lists some physical, mechanical, and thermal properties data collected from literature for sodium, one type of metal fuel (U-10Zr), and Type 316 stainless steel cladding used to construct the metal fuel pin. Among the three metals, sodium has the lowest melting point of only 98 °C [208 °F], which renders sodium the most susceptible to creep during transportation. For nuclear metal fuel, sodium does not provide a structural function and thus a stress would be applied by its own weight and by vibrations during transportation. As a rule of thumb, at temperatures below $0.4T_m$, where T_m is the melting point of the metal in Kelvin (K), thermal

Table 2-1. Physical, mechanical, and thermal properties of some materials			
Properties	**Na**	**U-10Zr**	**Type 316 stainless steel**
Density, g/cm^3	0.966 at 20 °C [68 °F]*	15.5†	8.0‡
Melting point, °C	98*	1,230†	1,427
Modulus of elasticity, GPa	10	179¶	203‡
Tensile strength, minimum, MPa	—	762¶	515‖
Yield strength, minimum, MPa	—	1,125¶	205‖
Thermal conductivity, W/m-K	130 at 20 °C [68 °F]*	15 at 20 °C [68 °F]¶	140 at 27 °C [81 °F]
Heat capacity, J/kg-K	1,230 at 20 °C [68 °F]*	260†	502 at 27 °C [81 °F]
Coefficient of thermal expansion, K^{-1}	7.1×10^{-5} at 25 °C [77 °F]	1.3×10^{-5} over 20–100 °C¶	1.8×10^{-5} at 25 °C [77 °F]**

*IAEA. "Liquid Metal Coolants for Fast Reactors Cooled by Sodium, Lead, and Lead-Bismuth Eutectic." International Atomic Energy Agency. Nuclear Energy Series No. NP-T-1.6. Vienna, Austria. 2012.
†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.
‡ASTM International. A240/A240M-19, "Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels and for General Applications." West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2019.
¶Oklo Inc. "Part II. Final Safety Analysis Report." Sunnyvale, California: Oklo Inc. 2020.
‖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.
**Leibowitz, L., E.C. Chang, M.G. Chasanov, R.L. Gibby, C. Kim, A.C. Millunzi, and D. Stahl. "Properties for Liquid Metal Fast Breeder Reactor Safety Analysis." ANL-CEN-RSD-76-1. Lemont, Illinois: Argonne National Laboratory. 1976.

activation is insufficient to produce creep in metals (Cadek, 1988). With a melting point of 371 K [98 °C], temperatures of at least 148 K [−125 °C] are required to initiate creep in sodium. The temperature ranges between −40 °C [−40 °F] and 38 °C [100 °F] per 10 CFR 71.71(c)(1,2) under NCT are well above 148 K [−125 °C]. As a result, sodium metal would creep under NCT, creating vacancies between sodium and cladding and metal fuel slugs such that the bonding between components of metal fuel pins could be compromised. Furthermore, the modulus of elasticity data in Table 2-1 shows that sodium is a very soft metal among the 3 metals. Under the influence of vibration and drop during NCT, sodium may shift in location. Information gap exists on sodium creep and location shift susceptibility and its effects on the geometric form of fresh nuclear metal fuel during NCT.

3) Tests and analysis under hypothetical accident conditions

Tests and analysis under HACs include: (i) 30 feet free drop to demonstrate that package damage is not significant and remains subcritical, and containment and shielding are maintained; (ii) crush; (iii) 40-inch puncture; (iv) 30-minute fire at 800 °C [1,472 °F]; and (v) immersion in water to demonstrate that the compressive yield stress, $\sigma_{y,}$ < 207 MPa.

The 30-feet free drop test precedes both puncture and fire tests and is designed to challenge fuel rod integrity, thermal protection, containment, shielding, and criticality control. As mentioned previously, sodium is a very soft metal. Under the influence of impact from 30-feet free drop, sodium may shift in location and potentially compromise the bonding achieved during fabrication. Furthermore, the cladding is welded to form a containment boundary for the fuel from loss or dispersal. Information gap exists on the ability of the metal fuel pins to withstand the specified drop condition and maintain containment and criticality functions.

The immersion test under HAC evaluates the effects of static water pressure head on the structural integrity of the package. The regulations described in 10 CFR 71.73(c)(6) state that the package must be immersed under a head of water of at least 50 ft [15 m] for at least 8 hours in the most damaging orientation. For some packages such as the Westinghouse Traveller package (Westinghouse Electric Company LLC, 2019) and RAJ-II (Global Nuclear Fuel, 2018) for fresh fuel transportation, the outerpack or outer container is not sealed to be leak-tight under external overpressure. During immersion test, water would fill the inner portion of the package applying hydrostatic pressure on the leak-tight fuel rod. The LWR fuel rod is usually able to withstand the hydrostatic pressure because it is designed to operate under pressurized condition during LWR operation. However, the sodium-cooled fast reactor is usually not operated under pressurized conditions and the metal fuel may not be designed to withstand the water pressure. If the water pressure compromises the cladding integrity, the inleakage of water could chemically react with sodium and fuel, which will significantly compromise the fuel properties. As a result, information gap exists on the ability of the cladding to withstand the increased external pressure from immersion test and its effects on fuel properties.

4) Materials evaluation

Materials evaluation includes: (i) mechanical and thermal properties of package materials; (ii) chemical interaction and galvanic coupling between materials under the influence of air or water or both; and (iii) cladding properties (e.g., total strain absorption energy, yield stress, ultimate stress). The commonly used packaging materials and metal fuel pin components including metal fuels, sodium, and cladding are discussed as follows.

Commonly used packaging materials: NRC certified a large number of transportation packages for shipping fresh LWR fuel, spent LWR fuel, and radioactive waste and a few are certified for shipping non-LWR fuel such as the Versa-Pac Models VP-55 and VP-110 for shipping fresh TRISO fuel and TN-FSV for shipping high-temperature gas-cooled reactor spent fuel (NRC, 2013). For these packages, Type 304 stainless steel is widely used to construct many safety-related package components for providing structural strength such as outerpack, outer shell, inner shell, bolts, and fasteners. Type 304 and Type 316 differ slightly in chemical composition (Type 316 stainless steel contains molybdenum for higher pitting corrosion resistance), but they belong to the same austenitic stainless steel group. Types 304 and 316 stainless steels are two of the six materials that have been accepted into ASME's Boiler and Pressure Vessel Code (B&PV) Section III, "Rules for Construction of Nuclear Facility Components," Division 5, "High Temperature Reactors" (Wright and Sham, 2018). Some physical, mechanical, and thermal properties of Type 304 stainless steel are almost the same as those for Type 316 stainless steel (ASTM International, 2019) as listed in Table 2-1a. Temperature-dependent material properties can be primarily obtained from Section II, Part D, of the ASME B&PV code (ASME, 2015). Because of the long and proven history in using Type 304 stainless steel by the industry, important properties of this material needed to evaluate its structural performance are expected to be available in the literature. As a result, no information gap is identified at the moment for Type 304 stainless steel material to be potentially used in transportation packages for fresh metal fuel.

In addition to Type 304 stainless steel, impact limiters such as polyurethane foam or wood or other materials encased in stainless steel (NRC, 2013) are also commonly used to construct many safety-related package components. Some important data used to evaluate the structural performance include (i) compressive or crush strength as a function of temperature, (ii) density, (iii) modulus, and (iv) Poisson's Ratio. Because these materials are commonly used, the information is expected to be available in the literature to evaluate their structural performance.

Other commonly used packaging materials include neutron absorbing and shielding materials. Neutron absorber materials such as Boral® are also commonly used in package for criticality control. Some important data including areal density, dimensions, and effectiveness testing data per the specifications are described in Electric Power Research Institute (EPRI, 2009) and ASTM E748-19 (ASTM International, 2019). Because neutron and gamma radiation emitted from the allowable fresh fuel contents is negligible in quantity, the transportation packages for fresh fuel usually don't contain shielding materials such as the Westinghouse Traveller package (Westinghouse Electric Company LLC).

Metal fuels: As discussed in Hall et al. (2019a), metal fuels include U, U-Pu, U-Fs (Fissium, Fs, is an alloy left by the reprocessing cycle from EBR-II operation containing 2.4 weight percent Mo, 1.9 weight percent Ru, 0.3 weight percent Rh, 0.2 weight percent Pd, 0.1 weight percent Zr, and 0.1 weight percent Nb), U-Zr, U-Mo, U-Pu-Zr. Important fuel properties relevant to fresh fuel transportation include U-235 enrichment, chemical composition, physical properties, sizes, and numbers of fuel rods/pins/assemblies, and maximum quantity of fuel loaded in a package (Hall et al., 2019b). For the license application of an advanced micro-reactor submitted by Oklo Inc. (2020) to NRC, U-10Zr is the metal fuel, which has been used in EBR-II operation from 1964 to 1994. Some physical and thermal properties for U-10Zr are shown in Table 2-1. Compared to stainless steel, U-10Zr has higher theoretical density of 15.5 g/cm^3. The thermal conductivity of metal fuel is high, at 32 W/m-K. The melting temperature for the U-10Zr fuel is high, at 1,230 °C. In addition to the technological importance of U-Zr in the past, in near-term, and in future applications, U-20Pu-10Zr was considered to be the reference metal fuel at the end of the research and development programs associated with both EBR-II and Fast Flux Test Facility (FFTF) in U.S. and is the startup metal fuel for the Versatile Test Reactor currently under design by the U.S. Department of Energy (DOE). It has also been recognized as an important metal fuel in the international Generation-IV Sodium Fast Reactor Program (Carmack et al., 2009).

Recently, Janney (2018a, b), Janney and Hayes (2018), Janney et al. (2020, 2019) complied and provided comprehensive review of available experimental data, worldwide, on fresh metal fuels based on U-Pu-Zr system (U-Zr, Pu-Zr, U-Pu, and U-Pu-Zr alloys, including those with minor actinides, rare-earth elements, and Y) and U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system. The available data are mainly on phases and phase diagrams, electrical properties, thermal expansion, thermal conductivity, heat capacity, mechanical properties, vapor pressures, and thermodynamic properties generated from experimental measurements before 1980. The authors (Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019) recognized the difficulty in taking extensive safety precautions in fabricating the fuel and obtaining reliable data on metal fuel properties for many reasons such as complex phase transformations, slow phase-transformation kinetics, sensitivity on sample preparation, rapid oxidation, and sensitivity to contaminants particularly oxygen, nitrogen, and carbon. Because of these difficulties, the authors observed that differences between measured values for alloys with the same nominal compositions from different measurements vary widely and documentation of sample compositions and experimental methods may be incomplete. According to Janney and Hayes (2018), some available data published before 1970 on U-10Zr were based on measurements from poorly characterized samples and the authors identified some disagreements among some data that are critical to characterize fuel properties. Janney et al. (2020) also commented that careful control of compositions including impurities, phases, and microstructures in U-Pu-Zr alloys is critical to obtain accurate data on thermal and mechanical properties. Based on the review by these researchers, information gaps exist on accurate data to characterize metal fuel properties including phases, phase diagrams, heat capacity, and mechanical properties. Knowledge of these properties may not be used to evaluate fuel performance during transportation, but it is important for understanding fuel constituent redistribution, fuel swelling and creep, fission gas release, melting or formation of liquid phases during normal reactor operations and transient reactor scenarios.

Sodium: The properties of sodium offer some advantages and disadvantages in using sodium in the fuel pin. Sodium has only one stable isotope, sodium-23, which is a very weak neutron absorber and weak neutron moderator. Another advantage of sodium is that it melts at 98 °C [208 °F] and boils at 883 °C [1,621 °F], a total temperature range of 785 °C [1,445 °F] between melting and boiling points. The wide temperature range enables the liquid phase to absorb significant heat. Furthermore, because the boiling point of sodium is higher than the reactor's operating temperature, sodium remains as liquid without leading to significant pressure increase in the fuel pin during reactor operation. Table 2-1 shows that sodium has high heat capacity and similar thermal conductivity as Type 316 stainless steel cladding. If there are no contaminants such as O_2 dissolved in sodium, pure sodium is not corrosive to stainless steel (Allen, 2019). One disadvantage is that sodium nuclei can absorb neutrons producing radioactive sodium-24; however, the activated sodium has a half-life of only 15 hours. An overwhelming disadvantage of sodium is its extremely high chemical reactivity with water, which produces sodium hydroxide and hydrogen, and the hydrogen burns when in contact with air. In the chemical industry transporting sodium, sodium is classified as a dangerous class Division 4.3 material (flammable substances in contact with water that emit flammable gases). Because of the lack of prior experience in transporting fresh sodium-containing metal fuel pin or assembly in the nuclear industry, information gap exists on safety protocols in transporting metal fuel in a transportation package for nuclear fuel, and effects of air and water on chemical interaction and galvanic coupling of package internal materials including sodium in the fuel pin.

Cladding: Cladding is a very important part of the fuel pin. It is expected to be designed to provide containment throughout the life of the fuel including transportation and while used in the reactor where it operates at much higher temperatures than transport conditions, and must contain the fuel, sodium, fission products, and gases. As mentioned in Hall et al. (2019), the metal fuel cladding materials used during EBR-II operation include Type 316 stainless steel, Alloy D9 (15Cr–15Ni–Mo–Ti austenitic stainless steel, a titanium modified variant of Type 316 stainless steel), and HT9 (Fe-12Cr-1Mo-0.5W-0.5Ni-0.25V-0.2C which is a high strength ferritic-martensitic stainless steel). Some physical and thermal properties for Type 316 stainless steel are shown in Table 2-1. Types 304 and 316 stainless steels were initially used especially in the U.S. because of their good long-term mechanical properties at high temperatures, chemical compatibility with sodium and stability in contact with uranium and plutonium-fuels. Alloy D9 was developed in the U.S. as an alternative stainless steel to overcome radiation-induced void swelling and creep of Types 304 and 316 stainless steels during reactor operation at some burnup levels [about 10 atomic percent (at.%)]. Figure 2-1 shows some examples of cladding materials swelling under the influence of irradiation from FFTF (Garner, 1993). Swelling of the cladding leads to dimensional expansion and bowing, which may indirectly lead to constriction of coolant flow, resulting in overheating and final failure due to either creep of cladding under fission gas pressure, fuel clad mechanical interaction or irradiation induced hardening. For the Oklo application (Oklo Inc., 2020), the reactor cell can is constructed using Type 316L stainless

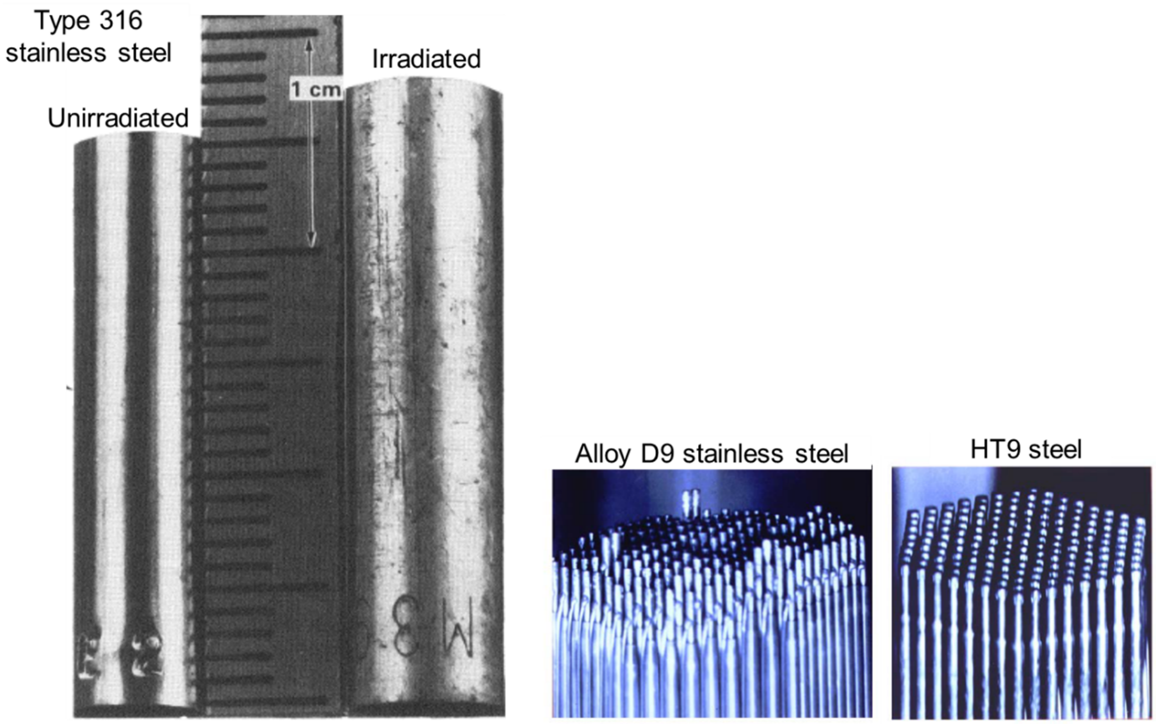


Figure 2-1. Examples of cladding materials swelling under the influence of irradiation from FFTF (Garner, 1993). The uneven length of fuel pins are due to swelling.

steel, which will be in contact with sodium at one side in the cell can, and with supercritical CO_2 on the other side. The selection of this material may be appropriate because the microreactor is only designed to operate at burnup less than 1 at.%. Different varieties of ferritic, ferritic-martensitic alloys including HT9, other materials such as 9Cr-2W steels were employed and considered for application as cladding materials in Europe, Japan, Korea, and the U.S. (IAEA, 2012a). HT-9 was extensively tested in the FFTF in the U.S. up to 19 at.% and was found that the extent of swelling was greatly reduced compared to Type 316 stainless steel and Alloy D9 as seen in Figure 2-1. However, HT-9 showed limited creep strength at high temperature {about 660 °C [1,220 °F]}, limited fracture toughness at low temperature, and susceptibility to fuel cladding chemical interaction. At enhanced burnup, the use of austenitic stainless steels such as Type 316 stainless steel and Alloy D9 and conventional ferritic and ferritic-martensitic steels as cladding material is still limited because of swelling at high temperature. The potential for further improvement of austenitic steels to resist swelling appears to be very small.

There is an international desire to improve nuclear energy efficiency by increasing fuel burnups, reduce waste stream, and enhance accident tolerance. Some countries are developing new fuel cladding materials to achieve these goals. Desired material properties for cladding include but are not limited to high corrosion resistance in contact with the coolant such as sodium, high mechanical strength, high thermal conductivity, low thermal expansion coefficient, high irradiation-induced swelling resistance, high helium-induced embrittlement resistance, high phase stability upon irradiation, and high irradiation-induced creep resistance. Achieving higher burnups requires significantly improved cladding materials to resist void swelling, embrittlement, and loss of high temperature strength. The most advanced class of alloys currently under development is oxide dispersion strengthened ferritic or ferritic-martensitic alloys with small particles such as nano-clusters of Y_2O_3 and/or TiO_2 distributed in the alloys, which are very stable up to very high operating temperatures (IAEA, 2012a). There are limited data on these advanced cladding materials. Information gap exists on advanced cladding material properties that can be used to achieve high burnup especially material performance data under the influence of irradiation, which is important to select the appropriate material for the desired performance.

Table 2-2 summarizes (i) information/data to be reviewed to evaluate structural integrity of transportation packages for fresh metal fuel, (ii) information availability, and (iii) potential information/data gaps.

2.1.2 Thermal evaluation

The objective of the thermal review is to evaluate the thermal performance of the package for NCT and HAC. Information to be reviewed includes the following:

1) Thermal design

Thermal design includes (i) insulation material; (ii) peak temperature of the package and its contents; (iii) allowable values or criteria for normal and accident conditions; and (iii) thermal properties of packaging component materials and fuel assembly including melting temperatures and service temperature ranges.

Ceramic fiber and foams such as polyurethane foam and phenolic foam are commonly used as thermal insulation material among the certified packages (NRC, 2013). The structural material such as Types 304 and 316 stainless steels also provides insulation function to some extent. Important properties in the evaluation of thermal performance include properties such as density, maximum use temperature, and thermal conductivity. Associated properties for commonly used insulation materials are expected to be available in the literature. Temperature-dependent material properties for structural components can be primarily obtained from Section II, Part D, of the ASME Boiler and Pressure Vessel (B&PV) Code. Table 2-1 in this report lists some physical, mechanical, and thermal properties of typical fuel pin components (Na, U-10Zr fuel, and Type 316 stainless steel) at certain temperatures. Many of these properties are temperature dependent such as the thermal conductivity of Type 316 stainless steel (Leibowitz et al., 1976) and U-10Zr (Janney, 2018) shown in Figure 2-2. Leibowitz et al. (1976) included additional temperature-dependent data on Types 304 and 316 stainless steels as commonly used structural materials. Additional data on thermal properties of metal fuel are available at Janney (2018a,b), Janney and Hayes (2018), Janney et al. (2020, 2019); however, as identified in Section 2.1.1, information gap exists on accurate data to characterize metal fuel properties including phases, phase diagrams, heat capacity, and mechanical properties. Peak temperature of the package and its contents and allowable values or criteria for normal and accident conditions would depend on the actual thermal design. Although no information on thermal design of transportation package for sodium-containing fresh metal fuel was found, information is expected to be available to make an assessment at the time of an application.

Table 2-2. Information to be reviewed and gaps for evaluating structural integrity of transportation packages for fresh metal fuel			
Areas of review	**Key information to be reviewed**	**Information availability**	**Potential information gaps**
Structural design	Description; Criteria; Weights and centers of gravity; General standards; Lifting and tie-down standards for all packages	Information available on many NRC-certified package designs (NRC, 2013), but not specifically for sodium-containing fresh metal fuel pins/assembly. Metal fuel design can vary.	Depending on metal fuel design at the time of an application
Tests and analysis under normal conditions of transport	Heat; Cold; Reduced external pressure; Increased external pressure; Vibration; Water spray; Free drop; Corner drop; Compression; Penetration	Some thermal and mechanical properties data are available for analysis, however, no tests and analysis currently available for sodium-containing fresh metal fuel pins/assembly	Sodium creep and location shift susceptibility and its effects on the geometric form of fresh nuclear metal fuel
Tests and analysis under hypothetical accident conditions	30 feet free drop; Crush; Puncture; Thermal; Immersion	Some thermal and mechanical properties data are available for analysis, however, no tests and analysis for sodium-containing fresh metal fuel pins/assembly	Ability of the metal fuel pins to withstand the specified drop condition and maintain containment and criticality functions; ability of the cladding to withstand the increased external pressure from immersion test and its effects on fuel properties
Materials evaluation	Mechanical and thermal properties of package materials; Chemical interaction and galvanic coupling between materials under the influence of air or water or both; Cladding properties e.g., total strain absorption energy, yield stress, ultimate stress	Commonly used packaging materials are known (NRC, 2013) and physical, thermal, and mechanical properties data are available (ASTM International, 2019; ASME, 2015; Wright and Sham, 2018; EPRI, 2009; IAEA, 2012a,b), some data are also available on metal fuel and cladding	Accurate data to characterize metal fuel properties including phases, phase diagrams, heat capacity, and mechanical properties; safety protocols in transporting metal fuel in a transportation package for nuclear fuel; effects of air and water on chemical interaction and

Table 2-2. Information to be reviewed and gaps for evaluating structural integrity of transportation packages for fresh metal fuel			
Areas of review	**Key information to be reviewed**	**Information availability**	**Potential information gaps**
		(Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019; Carmack et al., 2009; Garner, 1993)	galvanic coupling of package internal materials including sodium in the fuel pin; advanced cladding material properties that can be used to achieve high burnup especially material performance data under the influence of irradiation

ASTM International. A240/A240M-19, "Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels and for General Applications." West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2019.

ASME. "ASME Boiler and Pressure Vessel Code. Section II Materials. Part D." New York, New York: American Society of Mechanical Engineers. 2015.

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.

EPRI. "Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications," Report 1019110. Palo Alto, California: Electric Power Research Institute. 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. 2012a.

IAEA. "Liquid Metal Coolants for Fast Reactors Cooled by Sodium, Lead, and Lead-Bismuth Eutectic." Nuclear Energy Series No. NP-T-1.6. Vienna, Austria: International Atomic Energy Agency. 2012b.

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.

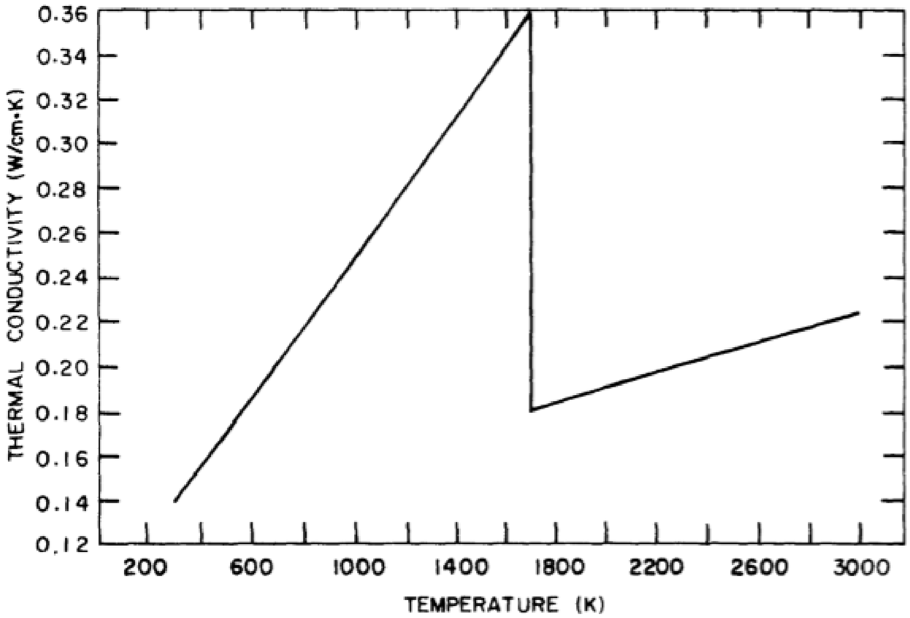
Janney, D.E., S.L. Hayes, and C.A. Adkins. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 2: Thermal and Mechanical Properties." *Nuclear Technology*. Vol. 206. pp.1–22. 2020.

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.

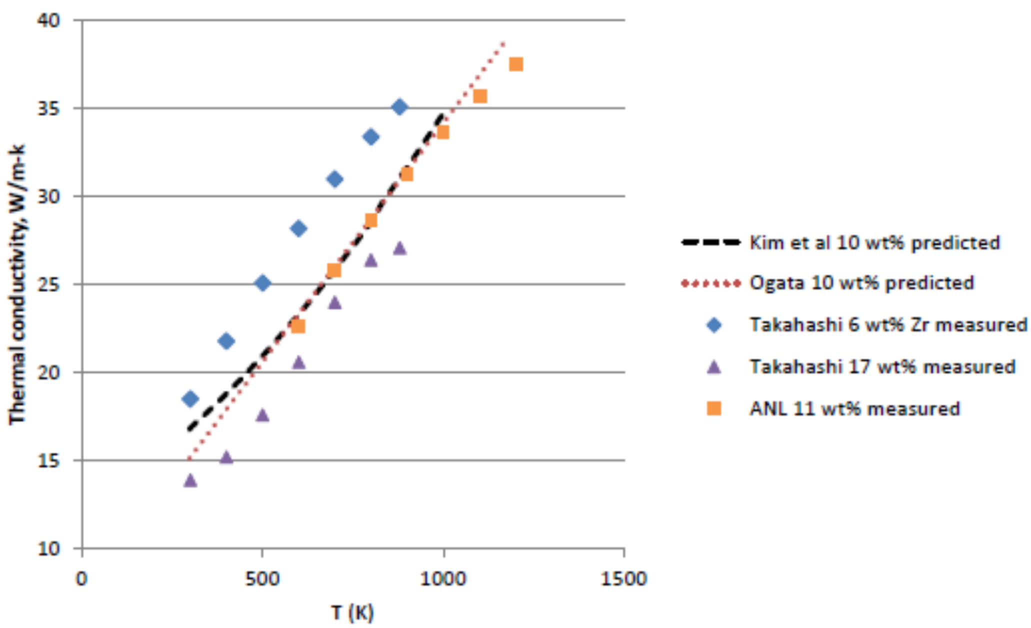
Janney, D.E. "Metallic Fuels Handbook, Part 2: Elements and Alloys not Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr." INL/EXT-15-36520 Revision 3 Part 2. Idaho Falls, Idaho: Idaho National Laboratory. 2018b.

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.

Wright, R.N. and S. Sham. "Status of Metallic Structural Materials for Molten Salt Reactors." INL/EXT-18-45171. Revision 0. 2018.



(a)



(b)

Figure 2-2. Thermal conductivity of (a) Type 316 stainless steel (Leibowitz et al., 1976) and (b) U-10Zr (Janney, 2018)

2) Thermal evaluation under NCT

Thermal evaluation under NCT includes (i) thermal analysis under heat and cold considering boundary conditions such as extreme ambient temperature and solar insolation to obtain data on maximum and minimum fuel and component temperatures, maximum normal operating pressure, and maximum and minimum pressures at the maximum and minimum temperatures, and (ii) calculations of differential thermal growth between material contacts based on thermal expansion coefficients.

Heat analysis considering boundary conditions includes an ambient temperature of 38 °C [100 °F] and solar insolation of 400 W/m² per 10 CFR 71.71(c)(1). Heat generation from fresh fuel contents is usually assumed to be insignificant. For cold conditions, the minimum environmental temperature that the package will be subjected to is −40 °C [−40 °F] per 10 CFR 71.71(c)(2). Pressure change in a contained environment is typically evaluated based on ideal gas law and temperature changes. Types 304 and 316 stainless steels are certified as high temperature structural component materials that can be used up to 650 °C (Wright and Sham, 2018), which is well above the maximum temperature under NCT. Table 2-1 shows that U-10Zr has a melting point of 1,230 °C [2,246 °F]and other metal fuels have melting points in similar range as U-10Zr, which are also well above the maximum temperature under NCT. Furthermore, stainless steels and metal fuels as metals are not sensitive to low temperature limit of −40°C [−40 °F] and are not sensitive to pressure limits under NCT. As a result, no information gap is expected for stainless steel and metal fuel for their thermal performance under NCT. Table 2-1 shows that the melting point of sodium (98 °C) is close to the extreme ambient temperature of 38 °C [100 °F]and the thermal expansion coefficient of sodium is about 4 times higher than Type 316 stainless steel and the metal fuel. In addition, sodium is a much softer metal compared to stainless steel and the metal fuel. This suggests that for the fuel pin components including cladding, sodium, and metal fuel, sodium would be more temperature and pressure sensitive. The influence of heat and cold during NCT could lead to differential thermal expansion and stresses for these components, but there is no gap to buffer the expansion. These factors could potentially compromise the bonding between sodium and cladding and sodium and metal fuel slug achieved during fuel fabrication. As such, information gap exists on the influence of extreme heat and cold and differential thermal growth on bonding between sodium and cladding and sodium and metal fuel slug.

3) Thermal evaluation under HAC

Thermal evaluation under accident (fire) conditions in accordance with 10 CFR 71.73 includes maximum temperatures and temperature distribution recorded from fire testing or from thermal modeling analysis. Fire testing usually follows free drop testing and puncture testing used for evaluating structural integrity and it is performed to demonstrate: (i) the package meets or exceeds regulatory requirements of 10 CFR 71; (ii) fuel assembly survives intact, without potential release of radioactivity; (iii) fuel assembly survives without cladding rupture caused by excessive temperatures; and (iv) other components survive intact. 10 CFR Part 71.73 (c)(4) requires that heat tests be conducted at 800 °C [1,472 °F] for 30 minutes to demonstrate that the package does not melt or run out and maximum stresses of materials are not exceeded when subjected to the thermal conditions associated with a fire accident. As discussed earlier, because of the low melting point of sodium and larger thermal expansion coefficient of sodium compared to stainless steel and metal fuel, information is needed to show that the temperature of the sodium during the hypothetical fire condition inside the fuel pin is below the melting point and the thermal stress does not compromise cladding integrity during a fire accident.

Table 2-3 summarizes (i) information/data to be reviewed to evaluate thermal performance of transportation packages for fresh metal fuel, (ii) information availability, and (iii) potential information/data gaps.

Table 2-3. Information to be reviewed and gaps for evaluating thermal performance of transportation packages for fresh metal fuel			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Thermal design	Insulation material; Peak temperature of the package and its contents; Allowable values or criteria for normal and accident conditions; Thermal properties of packaging components materials and fuel assembly including melting temperatures and service temperature ranges	Thermal properties for commonly used insulation materials are available; some thermal properties of fuel pin components including cladding, sodium, metal fuel are available (Leibowitz et al., 1976; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Accurate data to characterize metal fuel properties including phases, phase diagrams, heat capacity, and mechanical properties
Normal conditions	Heat and cold: boundary conditions (extreme ambient temperature, solar insolation) for thermal analysis to demonstrate: (i) maximum and minimum fuel and component temperatures, (ii) maximum normal operating pressure, (iii) maximum and minimum pressures at the maximum and minimum temperatures; Differential thermal expansion: calculations of differential thermal growth between material contacts based on thermal expansion coefficients	Some thermal properties including thermal expansion coefficients of fuel pin components are available (Leibowitz et al., 1976; Janney, 2018a, b; Janney and Hayes, 2018; Janney et al., 2020, 2019)	Influence of extreme heat and cold and differential thermal growth on bonding between sodium and cladding and sodium and metal fuel slug
Accident (fire) conditions	Maximum temperatures and temperature distribution recorded from fire testing or from thermal modeling analysis	Some thermal properties including thermal expansion coefficients of fuel pin components are available	During the hypothetical fire condition, temperature of the sodium inside the fuel pin and effect of thermal stress on cladding integrity

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.

2.1.3 Containment evaluation

The objective of the containment review is to evaluate the containment performance of the package for NCT and HAC. Information to be reviewed includes the following:

1) General considerations

General considerations include (i) configuration of containment boundary and (ii) generation of flammable gas. The configuration of containment boundary varies depending on the package design for the specific contents (e.g., the Versa-Pac Shipping Container) classified as a Type A Fissile package for transporting fresh TRISO fuel and other fuels is bolted on to close the content, but the package is not sealed to be airtight so that the internal pressure of the package is near atmospheric pressure under all transport conditions (Century Industries, 2010). Similarly, the outerpack or outer-container for the Westinghouse Traveller and RAJ-II shipping packages designed to transport Type AF and Type BF fresh fissile materials is closed but is not sealed to be airtight (Westinghouse Electric Company LLC., 2019; Global Nuclear Fuel, 2018). The containment boundary is the zirconium alloy cladding together with the welded end plugs of the fuel rods. For the Versa-Pac, Westinghouse Traveller, RAJ-II packages used to transport fresh fuel, no backfill with inert gas is applied to the package and any flammable gases, if generated, are not expected to accumulate because the packages are not sealed. Although no information on containment design and analyses of transportation package for sodium-containing fresh metal fuel was found, information is expected to be available to make an assessment at the time of an application.

2) Normal conditions

As mentioned earlier, metal fuel could be based on HALEU with enrichments ranging from 5 to 20 percent and a Type BF package may be required. For Type BF package, the applicant is expected to calculate the release rate to meet the requirements in 10 CFR 71.51 and then determine what leak rate is needed for the test. Leak rate tests for Type BF package containment boundary under NCT meeting ANSI N14.5-2014 “leaktight” criterion (ANSI, 2014) are often performed such as tests for the Westinghouse Traveller package (Westinghouse Electric Company LLC., 2019) according to Regulatory Guide 7.4 (NRC, 2019). ANSI N14.5-2014 defines leaktight as “A degree of package containment that in a practical sense precludes any significant release of radioactive materials. This degree of containment is achieved by demonstration of a leakage rate less than or equal to 1×10^{-7} reference·cm^3/s, of air at an upstream pressure of 1 atmosphere (atm) absolute (abs) and a downstream pressure of 0.01 atm abs or less.”

For the metal fuel pin, the cladding and welds are expected to be the containment boundary for the fuel and sodium inside the pin and fission gas during reactor operation no matter if the package is sealed to be airtight or not. For the fuel pin fabrication, welding of the end caps is a critical processing step, which is usually carried out by autogenous gas tungsten arc welding (Burkes et al., 2009). After welding, a leak-testing device is fit onto the weld and high-pressure helium is injected into the device to test if the weld is leak-proof by monitoring pressure change of the helium; (i.e., the weld leaks allowing helium in the device be forced into the fuel pin if there is a pressure decay) (Burkes et al., 2009). Different from ANSI N14.5-2014, this leak testing method was not governed by any standard and the leaktight criterion was not quantitatively defined. IAEA (2012b) reported that hot cracking is a major problem encountered during welding of austenitic stainless steel cladding due to the presence of impurities such as sulfur and phosphorous in the material and welding could also change the mechanical properties such as fracture toughness in the weld and heat affected zone. Information gap exists on fabrication standards for the fuel pin, data on cladding welds including total strain absorption energy, yield stress, ultimate stress as a function of temperature, release rate calculation , and criteria used to verify welds of the fuel pins integrity by non-destructive methods to ensure sufficient containment by the transportation package of the sodium-containing fuel under NCT for those certifications that rely on the integrity of the fuel to meet containment requirements.

3) Accident conditions

Evaluations for accident conditions include (i) release rate calculation; (ii) assessment of fuel rod integrity after thermal hypothetical accident conditions; and (iii) assessment of fuel rod content dispersal, crack, burst, buckling, and leaktight of cladding and welds from drop, fire, and other accident conditions. There are data in the literature and prior experience to demonstrate that LWR cladding together with welded end plugs can maintain its containment function during HAC such as the RAJ-II package (NRC, 2004; Global Nuclear Fuel, 2018), but no such data are available specifically for the metal fuel pin. Information gap exists on release rate calculation 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 under HAC for those certifications that rely on the integrity of the fuel to meet containment requirements.

Table 2-4 summarizes (i) information/data to be reviewed to evaluate containment performance of transportation packages for fresh metal fuel, (ii) information availability, and (iii) potential information/data gaps.

Table 2-4. Information to be reviewed and gaps for evaluating containment performance of transportation packages for fresh metal fuel

Areas of review	Information to be reviewed	Information availability	Potential information gaps
General considerations	Configuration of containment boundary; Generation of flammable gas	Configuration of containment boundary varies depending on the package design for the specific contents; package may be just closed, but not sealed to be airtight; cladding may be the only containment boundary (Westinghouse Electric Company LLC., 2019; Global Nuclear Fuel, 2018; Century Industries, 2010). Information for fresh metal fuel package is expected to be available to make an assessment at the time of an application.	None

Table 2-4. Information to be reviewed and gaps for evaluating containment performance of transportation packages for fresh metal fuel			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Normal conditions	Release rate calculation, containment boundary testing	ANSI N14.5-2014 (ANSI, 2014) and previous experience in leak testing of metal fuel pin (Burkes et al., 2009) are available	Fabrication standards for the fuel pin, data on cladding welds including total strain absorption energy, yield stress, ultimate stress as a function of temperature, release rate calculation , and criteria used to verify welds of the fuel pins integrity by non-destructive methods to ensure sufficient containment by the transportation package of the sodium-containing fuel under NCT for those certifications that rely on the integrity of the fuel to meet containment requirements.
Accident conditions	Release rate calculation; Fuel rod integrity after thermal hypothetical accident conditions; Fuel rod content dispersal, crack, burst, buckling, and leak-tight of cladding and welds from drop, fire, and other accident conditions	ANSI N14.5-2014 (ANSI, 2014) and previous experience in leak testing of metal fuel pin (Burkes et al., 2009) are available	Release rate calculation from drop, fire, and other accident conditions to ensure sufficient containment by the transportation package of the sodium-containing fuel under HAC for those certifications that rely on the integrity of the fuel to meet containment requirements
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. Global Nuclear Fuel. "RAJ-II Safety Analysis Report." Revision 10. ML18247A218. Wilmington, North Carolina: Global Nuclear Fuel-Americas, LLC. 2018. 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.			

2.1.4 Shielding evaluation

The objective of the shielding review is to demonstrate that the maximum allowable quantity of radioactive material will not result in exterior dose rates exceeding the established 10 CFR 71 limits and evaluate the shielding performance of the package under NCT and HAC. Information to be reviewed includes the following:

1) Package shielding design

Package shielding design includes (i) design features; and (ii) maximum radiation level from neutron and gamma depending on maximum quantity and enrichment of U-235 from dose rate analysis using codes such as Monte Carlo radiation transport code.

2) Radiation source

Radiation source includes content and quantity of radiological source terms (gamma and neutron).

3) Shielding model

Shielding model includes models for radiation level calculations to demonstrate maximum surface radiation level under normal and accident conditions.

4) Shielding evaluation

Shielding evaluation includes (i) methods; (ii) input and output data; (iii) flux-to-dose-rate conversion; and (iv) external radiation levels.

As previously mentioned, because neutron and gamma radiation emitted from the allowable fresh fuel contents is negligible in quantity, the transportation packages for fresh fuel usually do not contain shielding materials such as the Westinghouse Traveller package (Westinghouse Electric Company LLC, 2019) and shielding evaluation may not be performed such as the Versa-Pac Shipping Container (Century Industries, 2010). The necessity of shielding design for fresh metal fuel would depend on the source of the fuel. The radiological source terms for HALEU fuel derived from an irradiated uranium source such as EBR-II spent fuel are expected to be relatively more complicated. As the sources and methods for reprocessing uranium broaden and change to meet the anticipated increasing demand for HALEU fuel in the future for advanced reactors, the types and levels of impurities with residual radioactivity carried in the fresh fuel may become more variable. Some of the information in the literature regarding the sources of uranium and the different methods for reprocessing is not clearly documented. Shielding evaluations would need to appropriately specify gamma sources and their energies, which would be present due to the existence of fission products in the reprocessed uranium. Residual radioactivity in the fuel would impact both package shielding design, as well as radiation source specification. Because of these uncertainties, information gaps exist on radiation source and corresponding shielding design for fuels with diverse sources of uranium and fabricated with different reprocessing methods. Although there is no prior experience on fresh metal fuel transportation packages, related information on shielding model and shielding evaluation is expected to be available to make an assessment at the time of an application.

Table 2-5 summarizes (i) information/data needs to be reviewed for evaluating shielding performance of transportation packages for fresh metal fuel, (ii) information availability, and (iii) potential information/data gaps.

Table 2-5. Information to be reviewed and gaps for evaluating shielding performance of transportation packages for fresh metal fuel			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Package shielding design	Design features; Maximum radiation level	Packages for fresh fuel usually do not contain shielding materials such as the Westinghouse Traveller package (Westinghouse Electric Company LLC, 2019) and the Versa-Pac Shipping Container (Century Industries, 2010).	Shielding design for packaging fuels with diverse sources of uranium recovered from spent fuel and reprocessing methods
Radiation source	Content and quantity of radiological source terms (gamma and neutron)	Limited on HALEU fuel from reprocessed source	Source term specification including gamma sources and their energies (present due to fission products from reprocessed uranium)
Shielding model	Models for radiation level calculations demonstrating maximum surface radiation level under normal and accident conditions	Available for many NRC-certified packages (NRC, 2013) and Westinghouse Traveller package (Westinghouse Electric Company LLC, 2019). Although no prior experience for fresh metal fuel package, information is expected to be available to make an assessment at the time of an application.	None
Shielding evaluation	Methods; Input and output data; Flux-to-Dose-Rate conversion; External radiation levels	Available for many NRC-certified packages (NRC, 2013) and Westinghouse Traveller package (Westinghouse Electric Company LLC, 2019). Although no prior experience for fresh metal fuel package, information is expected to be available to make an assessment at the time of an application.	None
Century Industries. "Safety Analysis Report for the Century Industries Versa-Pac Shipping Container." Revision 3. ML101110135. Bristol, Virginia: Century Industries. 2010. 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.			

2.1.5 Criticality evaluation

The objective of the criticality review is to demonstrate that the transportation package design meets the nuclear criticality safety requirements in 10CFR 71.55 for a single package and 10CFR 71.59 for an array of packages under NCT and HAC. Information to be reviewed includes the following:

1) Criticality design

Criticality design includes (i) design features; (ii) maximum value of the effective multiplication factor; and (iii) criticality safety index.

2) Fissile material contents

Fissile material contents include content and type of fissile material.

3) Criticality model and evaluation

Criticality model and evaluation include (i) models for criticality evaluations demonstrating subcritical margins are maintained for single package and package arrays under NCT and HAC; (ii) material properties; (iii) computer code and cross-section library; and (iv) input data for criticality calculations.

4) Benchmark evaluation

Benchmark evaluation includes (i) applicability of benchmark experiments and (ii) bias determination.

Information on criticality design and criticality model and evaluation is available from many NRC-certified packages (NRC, 2013), although no information for sodium-containing fresh metal fuel was found. Such information and the fissile material contents for metal fuel are expected to be available to make an assessment at the time of an application. However, for nuclear metal fuel, typical enrichment levels ranged from 52 to 78 percent U-235 (Fast Reactor Working Group, 2018). Most available criticality benchmarking data correspond to less than 5 weight percent enriched or greater than 20 weight percent enriched uranium materials with a limited amount between 5 and 20 weight percent enriched (Jarrell, 2018). There are no benchmark experiments with compositions, configurations, and nuclear characteristics that are comparable to nuclear metal fuel. Because the computer codes used for criticality evaluations need to be benchmarked with sufficiently diverse data to verify predictions from the criticality analysis, an potential information gap exists on a potential lack of criticality benchmark data for transporting fresh metal fuel. As such, an assessment of criticality benchmark data and validation of existing criticality codes and methods is needed for transportation of fresh metal fuel.

Table 2-6 summarizes (i) information/data needs to be reviewed to evaluate criticality of transportation packages for fresh metal fuel, (ii) description of corresponding information/data that are available in the literature to support the reviews, and (iii) potential information/data gaps.

Table 2-6. Information to be reviewed and gaps for evaluating criticality performance of transportation packages for fresh metal fuel			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Criticality design	Design features; Maximum value of the effective multiplication factor; Criticality safety index	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
Fissile material contents	Content and type of fissile material	Information is expected to be available at the time of an application.	None
Criticality model and evaluation	Models for criticality evaluations demonstrating subcritical margins are maintained for single package and package arrays under normal conditions of transport and hypothetical accident conditions; Material properties; Computer code and cross-section library; Input data for criticality calculations	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
Benchmark evaluation	Applicability of benchmark experiments; Bias determination	Available for non-metal fuels with enrichments lower and higher than 5 weight percent U-235 (Jarrell, 2018)	A potential lack of criticality benchmark experiments for metal fuel packages

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.

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.

2.2 Fresh TRISO Fuel

TRISO fuel is a unique all-ceramic fuel form developed for high-temperature reactor designs, including HTGRs and fluoride salt-cooled high-temperature reactors (FHRs). To support commercialization of high-temperature reactor designs, EPRI led a collaborative effort involving DOE and industry to prepare a topical report, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance," (EPRI, 2019) that is under review and approval by NRC. This topical report consolidates data and analyses from the DOE-sponsored advanced gas reactor (AGR) program demonstrating the performance of UCO-based TRISO-coated particle fuel during irradiation and post-irradiation safety testing.

A number of TRISO fuel failure mechanisms have been identified under in-reactor or postulated accident conditions including (i) pressure vessel [i.e., the silicon carbide (SiC) layer] failure caused by internal gas pressure, (ii) irradiation-induced cracking and debonding of the pyrocarbon layers, (iii) fuel kernel migration, (iv) chemical attack of the SiC layer, (v) thermal decomposition of the SiC layer, and (vi) enhanced SiC permeability and/or SiC degradation (Hall et al., 2019c). These failure mechanisms are not expected to occur during transportation of fresh TRISO fuel due to the chemical characteristics of the fuel and lower stressors (e.g., deformation, temperature, radiation rate) in expected conditions during transport. This section provides more detailed information for potential safety reviews of transportation of fresh TRISO fuel within the context of the package safety evaluation topics of structural, thermal, containment, shielding, and criticality performance.

2.2.1 Structural evaluation

For packages certified by NRC for transportation of radioactive material, the packages must meet the structural requirements in §71.31, §71.33, §71.35, §71.71, and §71.73 under normal conditions of transport and hypothetical accident conditions. Under normal conditions of transport, the packages need to be tested under heat, cold, reduced external pressure, increased external pressure, vibration, water spray, free drop, corner drop, compression, and penetration conditions to ensure structural integrity. Under hypothetical accident conditions, the packages need to be tested under free drop, crush, puncture, thermal, and immersion conditions to ensure structural integrity. The structural design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999; SRP) are listed in Table 2-7.

The Versa-Pac package is certified to transport fresh TRISO fuel. The safety analysis report (SAR) for the Versa-Pac package contains a detailed description of the structural design and tests used to evaluate the package under the normal conditions of transport and the hypothetical accident conditions. Structural analyses of various Versa-Pac package components demonstrate that the package performance standards are satisfied (Century Industries, 2009). A detailed package structural design and structural analyses used to evaluate the package under normal conditions of transport and hypothetical accident conditions are expected to be described and evaluated by the applicant. As such, transportation of fresh TRISO fuel using existing certified transportation packaging is expected to provide adequate structural integrity for transporting this type of fuel.

Table 2-7. Information to be reviewed and gaps for evaluating structural integrity of transportation packages for fresh TRISO fuel

Areas of review	Information to be reviewed	Information availability	Potential information gaps
Structural design	Description; Criteria; Weights and centers of gravity; General standards; Lifting and tie-down standards for all packages	Existing NRC-certified Versa-Pac package provides all applicable structural design description; Detailed package structural design is expected to be described and evaluated by the applicant	None
Tests and analyses under normal conditions of transport	Heat and cold temperatures; Reduced and increased external pressure; Vibration; Water spray; 1 to 4 feet [0.3 to 1.2 m] free drop; Corner drop; Compression test; Penetration test	Existing NRC-certified Versa-Pac package provides all applicable structural evaluation under normal conditions of transport; Detailed tests and analyses used to evaluate the package under normal conditions of transport are expected to be developed and evaluated by the applicant	None
Tests and analyses under hypothetical accident conditions	30-foot [9.1 m] drop test; Crush test; 40-inch [1 m] puncture test; 30-minute fire at 1,475 °F [802 °C]; Immersion test	Existing NRC-certified Versa-Pac package provides all applicable structural evaluation under hypothetical accident conditions; Detailed tests and analyses used to evaluate the package under hypothetical accident conditions are expected to be developed and evaluated by the applicant	None
Materials evaluation	Mechanical and thermal properties of package materials; Chemical interaction and galvanic coupling between materials under the influence of air or water or both; Fuel coating layer properties	Existing NRC-certified Versa-Pac package provides all applicable materials evaluation; Detailed evaluation of package components and fuels is expected to be described and evaluated by the applicant	None

2.2.2 Thermal evaluation

For packaging certified by NRC for transportation of radioactive material, the design must meet the regulatory requirements applicable to the thermal evaluation in §71.31, §71.33, §71.35, §71.43, §71.51, §71.71, and §71.73 under normal conditions of transport and hypothetical accident conditions. The thermal design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999) are listed in Table 2-8.

The Versa-Pac package is certified to transport fresh TRISO fuel. The SAR for the Versa-Pac package contains a detailed description of the thermal evaluation with a maximum fuel temperature of 213 °F [100 °C] for normal conditions of transport and 291 °F [144 °C] for hypothetical accident conditions. Thermal analyses of the Versa-Pac package design demonstrate that the package performance in the area of thermal loading is satisfied (Century Industries, 2010). A detailed package thermal design and thermal analyses used to evaluate the package under normal conditions of transport and hypothetical accident conditions are expected to be described and evaluated by the applicant. As such, transportation of fresh TRISO fuel using existing certified transportation packaging is expected to meet the thermal performance requirements for transporting this type of fuel.

Table 2-8. Information to be reviewed and gaps for evaluating thermal performance of transportation packages for fresh TRISO fuel

Areas of review	Information to be reviewed	Information availability	Potential information gaps
Thermal design	Insulation material; Peak temperature of the package and its contents; Allowable values or criteria for normal and accident conditions; Thermal properties of packaging components, materials and fuel assembly including melting temperatures and service temperature ranges	Existing NRC-certified Versa-Pac package provides all applicable thermal design description; Detailed package thermal design is expected to be described and evaluated by the applicant	None
Thermal evaluation under normal conditions of transport	Heat and cold: boundary conditions (extreme ambient temperature, solar insolation) for thermal analysis to demonstrate: (i) maximum and minimum fuel and component temperatures and (ii) maximum and minimum pressures at the maximum and minimum temperatures; Differential thermal expansion: calculations of differential thermal growth between material contacts based on thermal expansion coefficients	Existing NRC-certified Versa-Pac package provides all applicable thermal evaluation under normal conditions of transport; Detailed package thermal evaluation under normal conditions of transport are expected to be developed and evaluated by the applicant	None

Table 2-8. Information to be reviewed and gaps for evaluating thermal performance of transportation packages for fresh TRISO fuel			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Thermal evaluation under hypothetical accident conditions	Maximum temperatures and temperature distribution recorded from fire testing or from thermal modeling analysis	Existing NRC-certified Versa-Pac package provides all applicable thermal evaluation under hypothetical accident conditions; Detailed package thermal evaluation under hypothetical accident conditions are expected to be developed and evaluated by the applicant	None

2.2.3 Containment evaluation

For packaging certified by NRC for transportation of radioactive material, the design must meet the regulatory requirements applicable to the containment evaluation in §71.31, §71.33, §71.35, §71.43, §71.51, §71.71, and §71.73 under normal conditions of transport and hypothetical accident conditions. The containment design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999) are listed in Table 2-9.

The Versa-Pac package is certified to transport fresh TRISO fuel. The SAR for the Versa-Pac package contains a detailed description and evaluation of the containment design. Performance tests, as required by §71.71 and §71.73, demonstrate that the Versa-Pac package design prevents loss or dispersal of the radioactive contents under the normal conditions of transport and the hypothetical accident conditions. Therefore, containment evaluations of the Versa-Pac package design demonstrate that the containment criteria are satisfied (Century Industries, 2010). A detailed package containment design and containment analyses used to evaluate the package under normal conditions of transport and hypothetical accident conditions are expected to be described and evaluated by the applicant. As such, transportation of fresh TRISO fuel using existing certified transportation packaging is expected to meet the containment requirements for transporting this type of fuel.

Table 2-9. Information to be reviewed and gaps for evaluating containment performance of transportation packages for fresh TRISO fuel			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Containment design	Configuration of containment boundary; Generation of flammable gas	Existing NRC-certified Versa-Pac package provides all applicable containment design description; Detailed package containment design is expected to be described and evaluated by the applicant	None
Containment evaluation under normal conditions of transport	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 Versa-Pac package provides all applicable containment evaluation under the normal conditions of transport; Detailed package containment evaluation under normal conditions of transport are expected to be developed and evaluated by the applicant	None
Containment evaluation under hypothetical accident conditions	Releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference air leakage rate calculated for hypothetical accident conditions in accordance with ANSI N14.5	Existing NRC-certified Versa-Pac package provides all applicable containment evaluation under hypothetical accident conditions; Detailed package containment evaluation under the hypothetical accident conditions are expected to be developed and evaluated by the applicant	None

2.2.4 Shielding evaluation

For packaging certified by NRC for transportation of radioactive material, the shielding design of the packages must meet the external radiation requirements in § 71.47 and § 71.51 under normal conditions of transport and hypothetical accident conditions. The shielding design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999) are listed in Table 2-10.

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). However, a detailed package shielding design and shielding analyses used to evaluate the package are expected to be described and evaluated by the applicant. As such, transportation of fresh TRISO fuel using existing NRC-certified transportation packaging is expected to meet the shielding safety requirements because of the low radiation of the fresh fuel.

TRISO fuel could be made from HALEU. A DOE-sponsored project is underway to design a new TRISO fuel fabrication facility. The project would use HALEU to produce the TRISO fuel pellets and pebbles for future high-temperature gas and molten salt reactors. Since there are various sources of uranium and reprocessing methods for providing HALEU feedstock with enrichments between 5 and 20 weight percent U-235, the types and levels of impurities residual radioactivity carried in the fresh fuel are uncertain. As a result, a potential information gap exists on radiation source for fuels with diverse sources of reprocessed uranium.

Table 2-10. Information to be reviewed and gaps for evaluating shielding performance of transportation packages for fresh TRISO fuel

Areas of review	Information to be reviewed	Information availability	Potential information gaps
Shielding design	Design features; Maximum radiation level	No shielding evaluation is performed for the existing NRC-certified Versa-Pac package because shielding is not required; Detailed package shielding design is expected to be described and evaluated by the applicant	Shielding design for packaging fuels with diverse sources of uranium and reprocessing methods
Radiation source	Content and quantity of radiological source terms (gamma and neutron)	No shielding evaluation is performed for the existing NRC-certified Versa-Pac package because shielding is not required; Detailed package radiation source is expected to be described and evaluated by the applicant	Source term specification including gamma sources and their energy, arising from fission products from reprocessed uranium or other impurities
Shielding model	Models for radiation level calculations demonstrating maximum surface radiation level under normal and accident conditions	No shielding evaluation is performed for the existing NRC-certified Versa-Pac package because shielding is not required; Detailed package shielding model is expected to be described and evaluated by the applicant	None
Shielding evaluation	Methods; Input and output data; Flux-to-Dose-Rate conversion; External radiation levels	No shielding evaluation is performed for the existing NRC-certified Versa-Pac package because shielding is not required; Detailed package shielding evaluation is expected to be developed and evaluated by the applicant meeting the shielding safety requirements due to the low radiation of the fresh TRISO fuel	None

2.2.5 Criticality evaluation

For packaging certified by NRC for transportation of radioactive material, the package design must meet the criticality safety requirements of § 71.55 for a single package and § 71.59 for an array of packages under normal conditions of transport and hypothetical accident conditions. The criticality design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999) are listed in Table 2-11.

The Versa-Pac package is certified to transport fresh TRISO fuel. The SAR for the Versa-Pac package contains a detailed description of the criticality design and models used for criticality evaluations, including single package, package arrays under normal conditions of transport, and package arrays under hypothetical accident conditions. Criticality evaluations of the Versa-Pac package design demonstrate that the criticality safety requirements are satisfied (Century Industries, 2010). A detailed package criticality design and criticality analyses for a single package and an array of packages under normal conditions of transport and hypothetical accident conditions are expected to be described and evaluated by the applicant.

As described in the SAR for the Versa-Pac package (Century Industries, 2010), the criticality safety analysis is based on 350 grams of 100 weight percent enriched U-235 as uranium metal, which bounds all other forms of uranium compounds. The computer code used for criticality calculations is benchmarked with critical experiments for enrichments ranging from 62.4 to 97.68 weight percent enriched U-235. A criticality safety analysis was also performed to support a high-capacity model to transport the maximum quantity of 695 grams of fissile material enriched up to 100 weight percent U-235 (NRC, 2016). The international consensus TRISO fuel design consists of high-density, low enriched UO_2 or UCO with a uranium enrichment less than 20 weight percent U-235 (INL, 2010). However, most available criticality benchmarking data correspond to less than 5 weight percent enriched or greater than 20 weight percent enriched uranium materials (Jarrell, 2018). No large-volume fresh TRISO fuel transportation packages have been certified by NRC. Since the computer codes used for criticality evaluations need to be benchmarked to verify predictions, there is a potential lack of criticality benchmark data for transporting fresh TRISO fuel in large quantities with enrichments in the range of 5 to 20 weight percent U-235. As such, an assessment of criticality benchmark data and validation of existing criticality codes and methods is needed for transportation of fresh TRISO fuel.

Table 2-11. Information to be reviewed and gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel

Areas of review	Information to be reviewed	Information availability	Potential information gaps
Criticality design	Design features; Maximum value of the effective multiplication factor; Criticality safety index	Existing NRC-certified Versa-Pac package provides all applicable criticality design description; Detailed criticality design is expected to be described and evaluated by the applicant	None

Table 2-11. Information to be reviewed and gaps for evaluating criticality performance of transportation packages for fresh TRISO fuel			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Fissile material contents	Content and type of fissile material	Existing NRC-certified Versa-Pac package provides all applicable specifications for the fissile material contents; Detailed fissile material contents are expected to be described and evaluated by the applicant	None
Criticality model and evaluation	Models for criticality evaluations demonstrating subcritical margins are maintained for single package and package arrays under normal conditions of transport and hypothetical accident conditions; Material properties; Computer code and cross-section library; Input data for criticality calculations	Existing NRC-certified Versa-Pac package provides all applicable considerations for criticality models and evaluations; Detailed criticality models and evaluations are expected to be developed and evaluated by the applicant	None
Benchmark evaluation	Applicability of benchmark experiments; Bias determination	Existing NRC-certified Versa-Pac package criticality analysis is based on 350 grams of 100 weight percent U-235; Benchmarking of the criticality codes against critical experiments for transporting fresh TRISO fuel in large quantities with enrichments 5 to 20 weight percent U-235 is not available	A potential lack of criticality benchmark data and applicability of existing criticality codes and methods

2.3 High-Assay Low-Enriched Uranium Hexafluoride (HALEU UF_6)

The proposed use of HALEU fuel has highlighted the need to consider issues important to the front end of the fuel cycle including transportation package certification for UF_6 enriched between 5 and 20 weight percent U-235. The production of HALEU in the form of UF_6 enriched between 5 and 20 weight percent U-235 would be transported and ultimately converted into other usable HALEU fuel forms including metals, oxides, or salts (Jarrell, 2018). Idaho National Laboratory has plans to recover and down blend HEU from several spent fuels, including EBR-II fuel, ATR fuel, and Naval Reactor fuel, using a process that involves separation of uranium from U-Zr alloys. Additionally, ongoing shipments of UF_6 with enrichments between 5 and 20 weight percent U-235 are anticipated to support planned HALEU production.

Physical and chemical characteristics or UF_6 are well documented. A typical phase diagram for UF_6 (Figure 2-3) indicates that it can exist in each or multiple phases; however, under normal transportation conditions (e.g., non-accident conditions) it is assumed to primarily exist as a solid, with the possibility, depending on environmental conditions, of minor phase change due to its low vapor pressure (IAEA, 1994).

In its solid state, UF_6 can be highly reactive under conditions in which it can behave as an oxidizing agent. An exothermic reaction of UF_6 could occur with water to form the soluble reaction products uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF) both of which are toxic (IAEA, 1994). For these reasons, rigorous procedures and quality assurance measures are implemented during package preparation, filling, and emptying for UF_6 cylinders. UF_6 obtained through reprocessing of irradiated uranium can, in addition to concentrations of U-234, U-235, and U-238, contain concentrations of other uranium isotopes such as U-232, U-233, U-236, and U-237, transuranic nuclides (e.g. Np-237, Pu-239), fission product impurities, and daughter products of these species (IAEA, 1994). The composition of reprocessed fuel will depend on reactor type, burnup, etc. Because chemical processes used for reprocessing do not affect the isotopic composition of uranium, the existence of impurities at reprocessing would remain though subsequent stages of conversion to UF_6 (IAEA, 1994). Standard UF_6 cylinders currently in service and approved for transportation are listed in Table 1 of ANSI N14.1 (ANSI, 2001).

Based on the UF_6 cylinder model information in ANSI N14.1 (ANSI, 2001, Table 1), cylinders are available for transportation of UF_6 up to 100 weight percent U-235, although fill limits are lower for UF_6 with higher enrichment values. UF_6 cylinders are normally shipped without protective overpacks, when the U-235 content does not exceed 1 weight percent U-235. For instances where quantities with U-235 greater than 1 weight percent U-235, overpack technology is typically used, and are either of the U.S. Department of Transportation (DOT) specifications 20PF or 21PF series, as prescribed in 49 CFR 178.120 and 178.121 of the DOT regulations, or as authorized in several NRC-issued CoCs. The NRC has specified the use of the ANSI N14.1 standard through the CoCs for fissile UF_6 transportation packages. Several CoCs for UF_6 overpacks using cylinders listed per ANSI N14.1 were identified in the NRC directory, NUREG-0383 (NRC, 20013, including the following:

- CoC No. 9196, Model UX-30, an overpack for 30-inch uranium hexafluoride (UF_6) cylinders, which is a right circular cylinder constructed of two stainless steel shells with the volume between the shells filled with 6-inch thick foam (7.8 - 9.8 PCF)

- CoC No 9362, DN30 package, a protective structural packaging (PSP) and the 30B uranium hexafluoride (UF_6) cylinder as specified in ANSI N14.1.

- CoC No. 6553, A protective overpack which provides impact and thermal resistance for the Model No. 48X 10-ton cylinder. Referred to as “Paducah Tiger”.

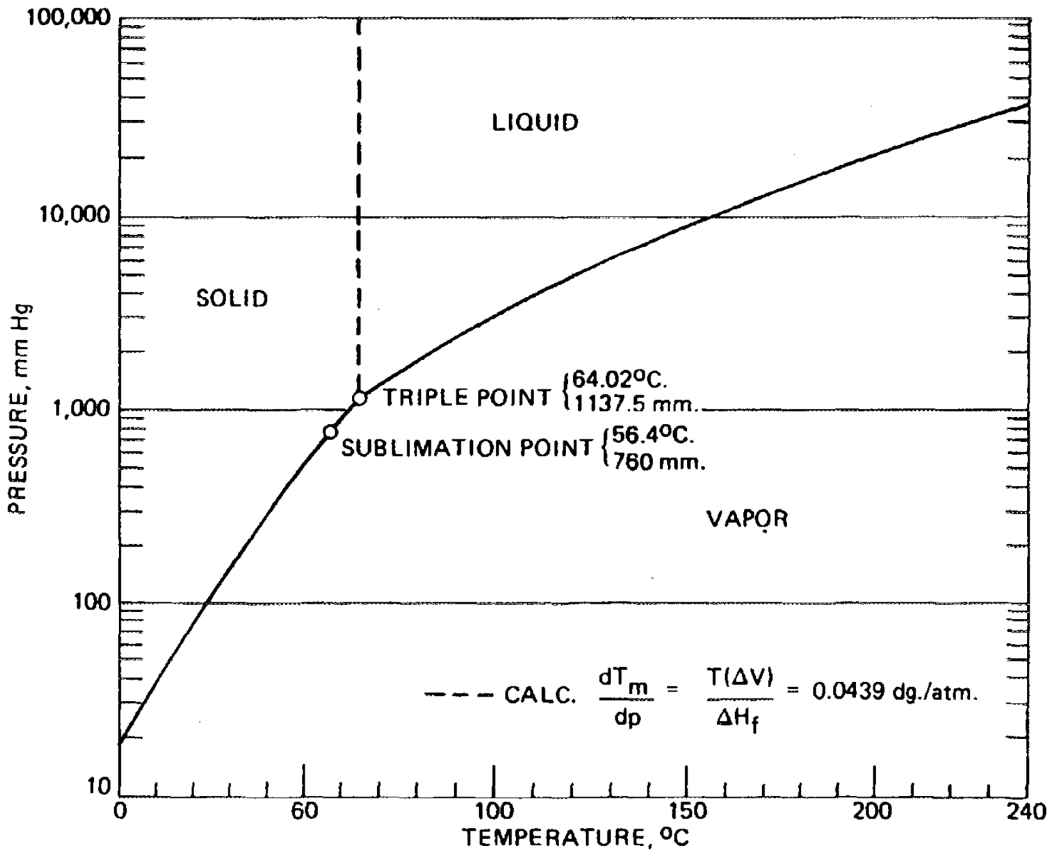


Figure 2-3. **UF_6 phase diagram (IAEA, 1994)**

DAHER Nuclear Technologies is developing packaging intended for the safe transport of UF_6 with enrichments of up to 20 weight percent U-235 without taking credit for moderator exclusion (PATRAM 2019). The package would include a new cylinder (called 30B-20) which is an overpack designed to accommodate a 30B cylinder that is currently listed in ANSI N14.1 and would accommodate up to 1,600 kg of UF_6 at 20 weight percent U-235. The package consists of both a standard 30-inch cylinder (e.g., such as the 30B, or equivalent), and the overpack packaging intended to protect the cylinder. The package would utilize similar structural packaging to the DN30 overpack model, and would possibly include a number of absorber rods, which would allow a much larger capacity of UF_6 per cylinder (PATRAM 2019).

2.3.1 Structural evaluation

Under normal conditions of transport (NCT), transportation packages are required to be tested under heat, cold, pressure, vibration, water spray, free drop to flat surface, corner drop, compression, and penetration scenarios to ensure structural integrity. Under hypothetical accident condition (HAC), the packages need to be tested under free drop, crush, puncture, thermal, and immersion conditions to ensure structural integrity. Structural design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999; SRP) are listed in Table 2-12. The UX-30 packaging is certified to transport unirradiated uranium, in the form of UF_6, with a U-235 mass percentage not to exceed 5 weight percent, and reprocessed uranium, in the form of UF_6, with a U-235 mass percentage not to exceed 5 weight percent, where the fission product gamma activity does not exceed 4.4×10^5 MeV Bq/kgU and the alpha activity from neptunium and plutonium is less than 3.3×10^3 Bq/kgU (CoC 9196). The safety analysis report (SAR) for the UX-30 packaging contains a description of the structural design and tests used to evaluate the package under the normal conditions of transport and the hypothetical accident conditions. Structural analyses of various UX-30 packaging components demonstrate that the package performance standards are satisfied (Columbiana Hi Tech, 2018). The UX-30 packaging approved capacity to up to 5,020 pounds of UF_6 with a U-235 isotope concentration of not more than 5 weight percent, which is a larger quantity of UF_6 proposed by the DAHER 30B-20 package, and as such the approach for structural analysis for NCT and HAC, in particular for drop test conditions would yield sufficient information to perform a structural analysis. Furthermore, a detailed package structural design and structural analyses used to evaluate the DAHER package under normal conditions of transport and hypothetical accident conditions are expected to be described and evaluated by the applicant. Based on the similarity of structural analysis performed for the UX-30 model, sufficient information is expected to be available to evaluate structural analysis for transportation packaging for UF_6 enriched to 20 weight percent U-235.

Table 2-12. Information to be reviewed and gaps for evaluating structural integrity of transportation packages for UF_6 enriched up to 20 weight percent U-235.

Areas of review	Information to be reviewed	Information availability	Potential information gaps
Structural design	Description; Criteria; Weights and centers of gravity; General standards; Lifting and tie-down standards for all packages	Existing NRC-certified UX-30 package provides a qualitative structural design description. Detailed package structural design is expected to be described and evaluated by the applicant	None
Tests and analyses under normal conditions of transport	Heat and cold temperatures; Reduced and increased external pressure; Vibration; Water spray; 1 to 4 feet [0.3 to 1.2 m] free drop; Corner drop; Compression test; Penetration test	Existing NRC-certified UX-30 package provides qualitative structural evaluations under normal conditions of transport. Detailed tests and analyses used to evaluate the package under normal conditions of transport are expected to be developed and evaluated by the applicant	None

Table 2-12. Information to be reviewed and gaps for evaluating structural integrity of transportation packages for UF_6 enriched up to 20 weight percent U-235.			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Tests and analyses under hypothetical accident conditions	30-foot [9.1 m] drop test; Crush test; 40-inch [1 m] puncture test; 30-minute fire at 1,475 °F [802 °C]; Immersion test	Existing NRC-certified UX-30 package provides a qualitative structural evaluation under hypothetical accident conditions. Detailed tests and analyses used to evaluate the package under hypothetical accident conditions are expected to be developed and evaluated by the applicant	None
Materials evaluation	Mechanical and thermal properties of package materials; Chemical interaction between materials under the influence of air or water or both, etc.	Existing NRC-certified UX-30 package provides similar materials evaluation. Detailed evaluation of package components and content is expected to be described and evaluated by the applicant	None

2.3.2 Thermal evaluation

For packaging certified by NRC for transportation of radioactive material, the design must meet the regulatory requirements applicable to the thermal evaluation under NCT and HAC as specified in 10 CFR Part 71. The thermal design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999) are listed in Table 2-13.

The UX-30 overpack has been tested for all NAC and HACs defined in 10 CFR Part 71 and demonstrate that the UX-30 package safety protects UF_6 cylinders (30A and 30B) from exceeding maximum design temperature. Testing to simulate NAC and HAC resulted in maximum temperature on the surface of the cylinder remaining below 93 °C [199 °F] (Columbiana Hi Tech, 2018). The maximum permissible temperature and pressure of the 30B cylinder is 121 °C [250 °F] and 200 psig, respectively, per ANSI 14.1 specifications; therefore, the overpack provided thermal protection well within the bounds of the 30B cylinder. A detailed DAHER 30B-20 package thermal design and thermal analyses used to evaluate the package under NCT and HAC are expected to be described and evaluated by the applicant. Based on the similarity of thermal evaluation performed for the UX-30 model, sufficient information is expected to be available to evaluate the thermal analysis for transportation packaging for UF_6 enriched to 20 weight percent U-235.

Table 2-13. Information to be reviewed and gaps for evaluating thermal performance of transportation packages for UF_6 enriched up to 20 weight percent U-235.			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Thermal design	Insulation material; Peak temperature of the package and its contents; Allowable values or criteria for normal and accident conditions; Thermal properties of packaging components, materials, and package contents including melting temperatures and service temperature ranges	Existing NRC-certified UX-30 package provides relevant applicable thermal design description. Detailed package thermal design is expected to be described and evaluated by the applicant	None
Thermal evaluation under normal conditions of transport	Heat and cold: boundary conditions (extreme ambient temperature, solar insolation) for thermal analysis to demonstrate: (i) maximum and minimum contents and component temperatures and (ii) maximum and minimum pressures at the maximum and minimum temperatures; Differential thermal expansion: calculations of differential thermal growth between material contacts based on thermal expansion coefficients	Existing NRC-certified UX-30 package provides relevant applicable thermal evaluation under normal conditions of transport. Detailed package thermal evaluation under normal conditions of transport are expected to be developed and evaluated by the applicant	None
Thermal evaluation under hypothetical accident conditions	Maximum temperatures and temperature distribution recorded from fire testing or from thermal modeling analysis	Existing NRC-certified UX-30 package provides relevant applicable thermal evaluation under hypothetical accident conditions. Detailed package thermal evaluation under hypothetical accident conditions are expected to be developed and evaluated by the applicant	None

2.3.3 Containment evaluation

The containment design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999) are listed in Table 2-14. As discussed in the UX-30 SAR, the UX-30 package is used in conjunction with a standard 30-inch cylinder such as the models 30B or 30C, as described in ANSI N14.1. The cylinder provides a containment boundary for the package. Design requirements per ANSI N14.1 includes internal pressure of 200 psig, an external pressure 25 psig, as well as internal temperature design limits between 4.4°C [−40 °F] to 121 °C [250 °F] . Under NCT, the 30B or 30C cylinder must have a leak rate of less than $1{\times}10^{-7}$ cm3 /sec, which was achieved per leak rate testing per ANSI N14.5. As described in the SAR, an analysis of leak rates determined that under NAC, a leak rate of less than $1{\times}10^{-7}$ cm3 /sec was achieved for the UX-30. A detailed DAHER 30B-20 package containment design and containment analyses used to evaluate the package under normal conditions of transport and hypothetical accident conditions are expected to be described and evaluated by the applicant. As such, sufficient information is expected to be available to evaluate a containment analysis for proposed overpack packaging for transportation of fresh UF_6 enriched to 20 weight percent U-235 using 30B cylinders.

Table 2-14. Information to be reviewed and gaps for evaluating containment performance of transportation packages for UF_6 enriched up to 20 weight percent U-235.

Areas of Review	Information to be Reviewed	Information availability	Potential Information gaps
Containment design	Configuration of containment boundary; Generation of flammable gas	Containment achieved via use of ANSI N14.1 30B cylinder, Existing NRC-certified UX-30 credits 30B cylinder. Additional detailed package containment design is expected to be described and evaluated by the applicant	None
Containment evaluation under normal conditions of transport	Containment boundary testing; For the leakage rate test of cylinders used for recycled UF_6, the cylinder must have a measured leak rate less than $1{\times}10^{-7}$ cm^3/sec	Containment achieved via use of ANSI N14.1 30B cylinder; existing NRC-certified UX-30 credits the 30B cylinder. Additional detailed package containment evaluation under normal conditions of transport are expected to be developed and evaluated by the applicant	None
Containment evaluation under hypothetical accident conditions	The package must have a measured leak rate less than $1{\times}10^{-7}$ cm^3 /sec. The internal pressure of a cylinder, under hypothetical accident conditions is dependent on the temperature of the UF_6 in the cylinder. The thermal analysis (see Section 3.5.3) shows most of the UF_6 is at 47.2 °C [117 °F] while a portion of the UF_6 can be assumed to be 93.3 °C [200 °F] or less	Containment achieved via use of ANSI N14.1 30B cylinder; existing NRC-certified UX-30 credits 30B cylinder. Detailed package containment evaluation under the hypothetical accident conditions are expected to be developed and evaluated by the applicant	None

2.3.4 Shielding evaluation

For packaging certified by NRC for transportation of radioactive material, the shielding design of the packages must meet the external radiation requirements in 10 CFR 71.47 and § 71.51 under normal conditions of transport and hypothetical accident conditions. The shielding design and evaluations reviewed by NRC using the NUREG–1609 standard review plan (NRC, 1999) are listed in Table 2-15. Per the SAR, UX-30 package shielding is sufficient to satisfy the dose rate limit of 10 CFR 71.51(a) (2) which states that any shielding loss resulting from the hypothetical accident will not increase the external dose rate to more than 1000 mrem/hr at one meter from the external surface of the cask. For the UX-30 shielding analysis, a source specification included 5,020 lbs of UF_6 enriched to 5 weight percent U-235, with the presence of a gamma source from fission products as a result of recycled UF_6. Under NCT, the UX-30 overpack is credited in the shielding evaluation; however, for HAC, the overpack is not credited. The external dose rates for the UX-30 package comply with the limits specified in 10 CFR 71.47 and §71.51 (Columbiana Hi Tech, 2018). A detailed package shielding design and shielding analyses used to evaluate the package are expected to be described and evaluated by the applicant. As such, for shielding evaluations for packages containing up to 20 weight percent U-235, an appropriate source term should be specified to bound the level of impurities that may exist in reprocessed UF_6, as well as all sources of gamma radiation. Provided sources of possible radiation from package contents can be fully characterized, sufficient information is expected to be available to evaluate a shielding analysis for transportation packaging for UF_6 enriched to 20 weight percent U-235.

Table 2-15. Information to be reviewed and gaps for evaluating shielding performance of transportation packages for UF_6 enriched up to 20 weight percent U-235.

Areas of review	Information to be reviewed	Information availability	Potential information gaps
Shielding design	Design features; Maximum radiation level	Shielding evaluation is performed for the existing NRC-certified UX-30 package; credit given to packaging under NCT. No credit provided to packaging under HAC. Detailed package shielding design is expected to be described and evaluated by the applicant	Shielding design for packaging fuels with variable sources of uranium and reprocessing methods

Table 2-15. Information to be reviewed and gaps for evaluating shielding performance of transportation packages for UF_6 enriched up to 20 weight percent U-235.			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Radiation source	Content and quantity of radiological source terms (gamma and neutron)	Shielding evaluation is performed for the existing NRC-certified UX-30 package. For this package, the fission products specified are primarily 144Ce, 134Cs, 137Cs, 95Nb, 103Ru, 106Ru, 99Tc and 95Zr (Columbiana Hi Tech, 2018). Detailed package radiation source is expected to be described and evaluated by the applicant.	Source term specification including gamma sources and their energies arising fission products from reprocessed uranium or radionuclide inventory in potential impurities that may exist from a previously irradiated U source.
Shielding model	Models for radiation level calculations demonstrating maximum surface radiation level under normal and accident conditions	For UX-30 package, shielding evaluation is performed using MicroShield, a point-kernel shielding code; Detailed package shielding model is expected to be described and evaluated by the applicant	None.
Shielding evaluation	Methods; Input and output data; Flux-to-Dose-Rate conversion; External radiation levels	For UX-30 package, shielding evaluation is performed using MicroShield, a point-kernel shielding code; Detailed package shielding evaluation is expected to be developed and evaluated by the applicant meeting the shielding safety requirements.	None

2.3.5 Criticality evaluation

For packaging certified by NRC for transportation of radioactive material, the package design must meet the criticality safety requirements of 10 CFR 71.55 for a single package and § 71.59 for an array of packages under normal conditions of transport and hypothetical accident conditions. The criticality design and evaluations reviewed by NRC using NUREG–1609 standard review plan (NRC, 1999) are listed in Table 2-16.

As discussed in the SAR for the UX-30 for criticality evaluation, the package relies on design features including the protective overpack, which prevents damage to the 30B cylinder sufficient to cause in-leakage of water during normal and accident conditions, and provides thermal protection which could prevent a release of contents as a result of fire (Columbiana Hi Tech, 2018). The UX-30 package also credits certain features on the 30C cylinder allowed per 10 CFR 71.55(c), such as the valve protective cover (VPC) for additional assurance against a failure to contain cylinder contents. Additionally, per the SAR, the UX 30 package relies on a directly applicable criticality safety evaluation performed by ORNL, which conservatively evaluates K_{eff} using an infinite array of packaging. ORNL performed criticality analysis for the 30B in 1991, which is the criticality analysis of record for the UX-30 overpack, Certificate of Compliance (CoC 9196) (PATRAM, 2019). As discussed in the criticality evaluation for the 30B cylinder performed by ORNL, a detailed criticality analysis determined that the maximum K_{eff} value for the conditions of optimal interstitial moderation with the premise of no water leakage into the UF_6 cylinders resulted in a K_{eff} less than the 0.95 upper subcritical limit criterion at all interstitial moderation conditions (ORNL, 1991). Additionally, the criticality evaluation considered a case of a 10-ton UF_6 cylinder with 5 wt percent U-235 enrichment, which satisfied criticality safety requirements of the 10 CFR Part 71 criteria for a Fissile Class I package without credit for the overpack. For transportation of UF_6 packages enriched up to 20 percent U-235, additional evaluations that demonstrate criticality safety may be necessary in order to address higher enrichments (e.g. up to 20 percent U-235) and mass (e.g. up to 1,600 kg) of the proposed DAHER 30B-20, with use of the 30B cylinders. Additionally, as discussed in the UX-30 SAR, recycled UF_6 can only be packaged using the packaging described in the SAR if the activity levels of the various isotopes contained in it do not exceed the A2 limits found in 49 CFR 173.433, which would ensure that requirements for purity control per ASTM C787 and C996 are met (Columbiana Hi Tech, 2018). For transportation of reprocessed UF_6, estimates of radionuclide compositions and their distribution throughout the package, would need to be specified and evaluated by the applicant as part of the criticality safety assessment for the proposed DAHER 30B-20 package. As discussed in the SAR, for the UX-30 package that uses a 30B cylinder, the computer code and cross-section validation consisted of determining K_{eff} for a series of 51 benchmark critical experiments. These benchmarks consisted of a full range of possible experiments including 11 highly enriched cases and 40 low-enriched cases. A criticality evaluation was also conducted to support a 14-ton cylinder though it was reviewed assuming a maximum U-235 enrichment of 5 percent. No transportation packages for larger volumes of higher enrichment UF_6 are currently approved, as the total UF_6 quantity listed for transport using the standard cylinders decreases with increasing enrichment above 5 weight percent U-235. A number of criticality experiments that are applicable to transportation packages with HALEU fuel have been identified; however, questions regarding specific package configuration, size, and contents remain (Jarrell, 2018). An assessment of the availability of criticality experiments for HALEU assays that consider larger, and potentially more reactive transportation package configurations with contents up to 20 weight percent U-235 would be necessary to determine the applicability of existing benchmark data used for criticality evaluations.

Table 2-16. Information to be reviewed and gaps for evaluating criticality performance of transportation packages for UF_6 enriched up to 20 weight percent U-235.			
Areas of review	**Information to be reviewed**	**Information availability**	**Potential information gaps**
Criticality design	Design features; Maximum value of the effective multiplication factor; Criticality safety index	Existing NRC-certified UX-30 package provides all applicable criticality design description. Detailed criticality design is expected to be described and evaluated by the applicant.	None
Fissile material contents	Content and type of fissile material	Existing NRC-certified UX-30 package provides all applicable specifications for the fissile material contents. Detailed fissile material contents (including limits for impurities) are expected to be described and evaluated by the applicant.	None
Criticality model and evaluation	Models for criticality evaluations demonstrating subcritical margins are maintained for single package and package arrays under normal conditions of transport and hypothetical accident conditions; Material properties; Computer code and cross-section library; Input data for criticality calculations	Existing NRC-certified UX-30 package provides all applicable considerations for criticality models and evaluations. Detailed criticality models and evaluations are expected to be developed and evaluated by the applicant	None
Benchmark evaluation	Applicability of benchmark experiments; Bias determination	Existing NRC-certified UX-30 package criticality analysis is based UF_6 cylinder with 5 weight percent U-235 enrichment. Benchmarking of the criticality codes against available critical experiments for transporting HALEU UF_6 assays in large quantities with enrichments 5 to 20 weight percent U-235 is limited.	A potential lack of criticality benchmark experiments for larger, more reactive configurations of UF_6.

3 INFORMATION NEEDS ASSOCIATED WITH TRANSPORTATION OF FRESH ARF TYPES

An earlier report in this series (Hall et al., 2019b) identified some potential challenges associated with transportation of fresh (unirradiated) advanced reactor fuel (ARF). These challenges were identified by comparing attributes of package designs approved by U.S. Nuclear Regulatory Commission (NRC) to characteristics of non-light water reactor (LWR) fuel expected to meet the safety requirements in 10 of the *Code of Federal Regulations* (10 CFR) Part 71. Some of the information gaps identified in Section 2 overlap with these challenges. Table 3-1 provides an overview of information gaps associated with transportation of fresh ARF and potential information needs to fill the corresponding gaps based on topics addressed by NRC regulations and guidance. The information needs are discussed in more detail in the following subsections.

Table 3-1. Summary of potential information gaps and information needs associated with transportation of fresh ARF types

Potential information gaps	Safety relevance	Potential changes to NRC regulations and guidance	Potential information needs based on potential Information gaps
Lack of criticality benchmark data and applicability of existing criticality codes and methods for ARF types with higher enrichment	Criticality	Supplemental review guidance related to criticality benchmarking	Criticality benchmarking
Shielding design for packaging fuels with uncertain sources of uranium and uncertain reprocessing methods	Shielding	Supplementary review guidance related shielding evaluation for fresh fuel containing recycled (irradiated) uranium.	Necessary properties to characterize radiation sources for fresh fuel containing recycled (irradiated) uranium, including burnup, radionuclide inventory, gamma sources and energy from these sources
Source term specification including gamma sources and energy from these sources that would be present due to fission products from reprocessed uranium or other impurities that may exist	Shielding	Supplementary review guidance related to shielding evaluation for fresh fuel containing recycled (irradiated) uranium.	Necessary properties to characterize radiation sources for fresh fuel containing recycled (irradiated) or reprocessed uranium, including burnup, radionuclide inventory, gamma sources and energy from these sources

Table 3-1. Summary of potential information gaps and information needs associated with transportation of fresh ARF types			
Potential information gaps	**Safety relevance**	**Potential changes to NRC regulations and guidance**	**Potential information needs based on potential Information gaps**
Structural integrity of metal fuel	Structural integrity	Supplementary review guidance related to fuel pins with bonding-sensitive subcomponent	Well-controlled experimental testing and modeling to obtain accurate and reliable data on metal fuel properties and cladding material physical, thermal, and mechanical properties; Establishment of codes and standards for any new materials and structures; Structural analysis of the fuel assembly given stresses the fuel assembly will experience during NCT and HAC
Containment of sodium-bearing metal fuel	Containment	10 CFR 71.43(d) (A package must be made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction among the packaging components, among package contents, or between the packaging components and the package contents, including possible reaction resulting from inleakage of water, to the maximum credible extent) Supplementary review guidance related to package containing highly moisture sensitive and pyrophoric material	Establishment of maximum allowable leakage rate; Design analysis, codes and standards, and confirmatory testing
Thermal performance of sodium-bearing metal fuel	Thermal	10 CFR 71.75(a)(3) (The specimen may not melt or disperse when subjected to the heat test —800 °C for 10 minutes) Supplementary review guidance related to fuel pins with low melting point subcomponent (98 °C for sodium)	Properties of any new materials or structures important to thermal analysis; Thermal analysis to model the temperature distribution; Confirmatory testing

3.1 Criticality Benchmarking

NRC regulations in 10 CFR Part 71 establish criticality requirements for transportation package approval. The packaging must be designed, and the contents specified, such that the package remains subcritical. Criticality safety evaluations require that criticality codes and methods are benchmarked against critical experiments that are applicable to the actual packaging design and contents. The NRC guidance on benchmarking of criticality codes is provided in the standard review plan, NUREG-1609 (NRC, 1999), and the draft standard review plan for comment, NUREG-2216 (NRC, 2019). NUREG–2216 (NRC, 2019) states that the benchmark experiments should have, to the maximum extent possible, the same materials, neutron spectrum, and configurations as the package evaluations for each type of contents. The benchmark evaluations include a comparison of the calculated and experimental results to determine the bias and the uncertainty associated with the bias. Subcriticality is assured if the sum of the effective neutron multiplication factor (k_{eff}), within two standard deviations, and all biases and bias uncertainties (i.e., the upper subcritical limit) is demonstrated to be less than 0.95. The industry guidance regarding criticality benchmarking and validation methods is provided in American Nuclear Society (ANS) Standard ANSI/ANS-8.1, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" (ANS, 2007). Additional information on benchmarking criticality evaluations are also discussed in several NRC documents including NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages" (NRC, 1997) and NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (NRC, 2001).

Given the potential lack of criticality benchmark data for transporting both fresh metal and tristructural isotropic (TRISO) fuels and UF_6 in large quantities with enrichments in the range of 5 to 20 weight percent U-235, criticality benchmark evaluations will either need to (i) develop new critical experiments with applicable materials and enrichments or (ii) extrapolate the bias and bias uncertainty beyond the range of applicability for which critical experiments exist. NUREG–2216 (NRC, 2019) states that for cases where extrapolation is necessary or data in the range of applicability are limited, additional margin on subcriticality should be considered in the analyses or uncertainty in the bias. There are guidance documents on the use of the bias and uncertainty for the evaluation of a package with characteristics beyond the range of applicability, including NUREG/CR-6698, NUREG/CR-5661, and ANSI/ANS-8.1 (NRC, 2001, 1997; ANS, 2007). An extrapolation of the range of applicability should be supported by sensitivity and uncertainty analyses in which only the parameters being extrapolated are varied to identify trends in the bias. A detailed technical basis must be provided in the absence of suitable critical experiments. The basis should either support the methods used for extrapolating the range of applicability or include additional margin to account for the increased bias uncertainty (NRC, 2019). However, a lack of applicable criticality benchmark data could lead to unfavorably conservative safety margins in criticality evaluations of transportation package designs for fresh metal and TRISO fuels.

For UF_6 limited criticality experiments for high-essay low-enriched uranium (HALEU) assays that consider realistic transportation package configurations with contents to 20 weight percent U-235 exist. Oak Ridge National Laboratory (ORNL) conducted criticality evaluation on the 30B cylinder which is relied upon as the criticality evaluation of record for the UX-30 package. Using the ORNL analysis, additional evaluations were conducted to understand the sensitivity of additional factors that contribute towards moderation such as cylinder wall thicknesses (e.g., manufacturing tolerances), as well as the worth of fluorine and iron, when an increased enrichment 6 weight percent U-235 was considered. The results indicated limitations in capacity and stacking of cylinders, without an overpack, as well as a sensitivity to different parameter that affect moderation (PATRAM, 2019). Additionally, the presence of volatile impurities such as hydrofluoric acid, and possibly inhomogeneity of density and voiding within the fissile contents, and presence of other impurities and residues (hydrated uranium residue) impact assumptions in criticality evaluations. Additionally, no larger package configurations for UF_6 that include sources of uranium from recycled or reprocessed spent nuclear fuel have been certified by NRC. As such, an assessment of the availability criticality benchmark data for these configurations and contents would reduce the possible need for conservative assumptions or safety margins in transportation package designs.

3.2 Radiation Sources and Shielding Design

As discussed in section 2.3, UF_6 obtained through processing of recycled spent (irradiated) fuel could contain concentrations of U-234, U-235, and U-238, other uranium isotopes such as U-232, U-233, U-236, and U-237, transuranic nuclides (e.g. Np-237, Pu-239), fission product impurities, and daughter products of these species. The level of residual radioactivity arising from the proposed processes for uranium recovery from spent fuel (e.g., Zircex or electrochemical process) and subsequent down-blending to HALEU is uncertain. As discussed in Sections 2.1.4 and 2.2.4, nuclear metal fuel and TRISO fuel fabricated with reprocessed uranium could possibly contain radioactive sources that should be considered in the shielding evaluation. Radionuclide inventories in possible impurities arising from previous irradiated uranium sources should be considered to specify the source term in shielding evaluations, or conservative shielding analysis would have to be implemented bounding source terms.

3.3 Structural Integrity of Metal Fuel

Table 2-2 identified some potential information gaps related to fresh metal fuel and advanced cladding material properties and structural integrity of cladding during NCT and hypothetical accident conditions (HAC). Potential information needs to address the corresponding gaps are identified as follows:

- Well controlled experimental testing and modeling to obtain accurate and reliable data on metal fuel properties and cladding material physical, thermal, and mechanical properties
    - Mechanical properties typically include tensile properties and fracture resistance
    - Mechanical properties need to account for environmental and operating conditions during NCT (hot and cold temperatures) and HAC, considering also the potential for microstructural changes at elevated temperatures
- Establishment of codes and standards for any new materials and structures included in package and welding criteria for containment-related, criticality-related, and other safety-related welds
- Determination of allowable stress values used for analytic assessments of package structural performance
- Structural analysis of the fuel assembly given stresses the fuel assembly will experience during NCT and HAC to demonstrate that the fuel has adequate structural integrity to satisfy the containment, shielding, and subcriticality requirements of 10 CFR Part 71. Analysis and confirmatory testing would aid in characterizing the relationship between stress and cladding integrity and demonstrate that the tests under NCT would not affect the package's ability to withstand the HAC tests.

3.4 Containment of Metal Fuel

Table 2-4 identified some potential information gaps related to containment of fresh metal fuel during NCT and HAC. Potential information needs to address the corresponding gaps are identified as follows:

- Establishment of maximum allowable leakage rate for normal conditions and demonstration that the packaging meets the maximum allowable leakage rate

- Design analysis, codes and standards, and confirmatory testing of the package containment boundary under normal and hypothetical accident conditions to demonstrate that there will not be inleakage of water

- Measures to ensure no failure of containment boundary that would lead to violent reaction of sodium with inleakage of water producing combustible gas

- Analysis to demonstrate correlation of any possible containment boundary leakage rate on chemical reactions of sodium and fuel with moisture and air

3.5 Thermal Performance of Metal Fuel

Table 2-3 identified some potential information gaps related to thermal performance of fresh metal fuel during NCT and HAC. Potential information needs to address the corresponding gaps are identified as follows:

- Properties of any new materials or structures important to thermal analysis. Thermal properties include thermal conductivity, thermal expansion, specific heat, density, heat capacity and others as a function of temperature over the ranges the components experience under the conditions associated with NCT and HAC tests

- Thermal analysis to model the temperature distribution to demonstrate that sodium during the hypothetical fire condition inside the fuel pin remains below the melting point, the thermal stress of the cladding does not exceed the limit, the cladding of the fuel, the thermal bond, and fuel is not stressed due to differential thermal expansion, the package has adequate thermal performance to meet the containment, shielding, subcriticality, and temperature requirements of 10 CFR Part 71

- Confirmatory testing to demonstrate that the geometric form of fuel would not be substantially altered and the maximum temperatures and pressures do not exceed the maximum allowable temperature limits

4 SUMMARY AND CONCLUSIONS

This report identifies technical information availability and potential information gaps to be addressed before or during licensing reviews of transportation packages intended for use with fresh advanced reactor fuel (ARF) types. The Center for Nuclear Waste Regulatory Analyses (CNWRA®) reviewed publicly available literature related to proposed ARF types and similar non-light water reactor (LWR) fuel, as well as applicable U.S. Nuclear Regulatory Commission (NRC) regulations and guidance that span the review topics outlined in NUREG–1609, to establish potential information gaps for the fuel types considered. ARF types considered for this report were nuclear metal fuel and tristructural isotropic (TRISO) fuel, along with fuel precursor UF_6 with possible enrichment up to 20 weight percent U-235. Literature was reviewed for the fuel types and the precursor UF_6 material to identify information gaps as they relate to transportation package structural integrity, thermal, containment, shielding, and criticality evaluations under normal and accident conditions, for enrichments up to 20 weight percent U-235.

TRISO fuel is being considered for use with a variety of advanced reactor designs, including high-temperature gas-cooled reactors and fluoride salt-cooled high-temperature reactors. The Versa-Pac is certified by NRC to transport fresh TRISO fuel. The Versa-Pac SAR evaluates and documents all applicable evaluations for structural, containment, thermal, shielding, and criticality analysis for normal and accident conditions. Because information for each evaluation required by 10 CFR Part 71 is complete and well documented in the Versa-Pac SAR, and throughout literature available on TRISO fuel, limited specific information gaps were identified for transportation package certification reviews for fresh TRISO fuel. However, the existing NRC-certified Versa-Pac package criticality analysis is based on 350 grams of 100 weight percent U-235, and criticality experiments for transporting fresh TRISO fuel in large quantities with enrichments 5 to 20 weight percent U-235 is limited. Therefore, additional criticality benchmark evaluations for large quantities of TRISO fuel enriched to 20 weight percent U-235 are necessary to demonstrate that the calculational method used to establish criticality safety has been validated against critical experiments.

Fresh metal fuel necessitates special consideration for transportation package safety review topics. Although some characteristics of metal fuel, including thermal and mechanical properties, are well documented, testing or analyses that document performance of sodium-containing fresh metal fuel pins/assemblies during normal and accident conditions, needed to support structural evaluations, are limited. Structural performance of a proposed transportation package would consider all degradation mechanisms applicable for proposed metallic fuel contents. Thermal properties of materials in metallic fuel are available, but accurate data to characterize phases, heat capacity, and mechanical properties are not well documented and are limited. This information is important for evaluating the influence of extreme temperature conditions on thermal growth and bonding for fuel assemblies at the region of connection between sodium and cladding, and sodium and the metal fuel slug. Documentation of fabrication standards for the proposed metal fuel pin, design information related to cladding welds, total strain absorption energy, yield stress, ultimate stress as a function of temperature, release rate calculation, and criteria used to verify fuel pin weld integrity by non-destructive methods to ensure containment of the sodium-containing fuel under NCT is limited and would be needed to adequately evaluate package performance against 10 CFR Part 71 requirements and other containment requirements. Additionally, if metal fuel is fabricated from high-essay low-enriched uranium (HALEU) fuel forms that contain reprocessed uranium containing specific radiological source terms, and other radiation sources, impurities, etc., a full assessment of these contents would be needed in order to properly perform shielding evaluations for fresh metal fuel transportation packages. Furthermore, an assessment of criticality data for fresh metal fuel is needed, which would be used to possibly reduce uncertainty in criticality evaluations or eliminate conservatisms in transportation package design for fresh metal fuel.

The production of HALEU in the form of UF_6 enriched between 5 and 20 weight percent U-235 would be transported and ultimately converted into other usable HALEU fuel forms, including metals, oxides, or salts. UF_6 can be highly reactive and form the toxic soluble reaction products uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF). UF_6 fabricated as part of the reprocessing of irradiated uranium can contain concentrations of uranium isotopes, transuranic nuclides, fission product impurities, and daughter products of these species. Chemical processes used for reprocessing do not affect the isotopic composition of uranium, and the existence of impurities at reprocessing would remain through subsequent stages of conversion to UF_6. Standard cylinders listed for transportation of UF_6 are listed in ANSI N14.1, including the 30B cylinder, a widely used steel cylinder approved for carrying UF_6 up to 5,020 lbs [2,277 Kg] weight percent U-235. Currently, no transportation packages for larger quantities of higher enriched UF_6 (e.g., up to 20 percent weight U-235) are approved by the NRC. DAHER Nuclear Technologies is developing packaging intended for the safe transport of up to 3,527 lbs [1,600 kg] of UF_6 with enrichments of up to 20 weight percent U-235 called 30B-20, which utilizes the 30B cylinder listed in ANSI N14.1. The UX-30 overpack, an NRC-approved overpack (CoC No. 9196) that also utilizes the 30B cylinder, provides a qualitative set of information that was reviewed across the topics in SRP1609, and documented in the UX-30 CoC and SAR. Of the review topics, design and regulatory requirements, and information related to structural, containment, and thermal evaluations were available and well documented; therefore, limited specific information gaps were identified related to the SRP topics for transportation of HALEU UF_6. The shielding evaluation in the UX-30 SAR specified fission products that would be considered in a shielding analysis, as well as source term specifications including gamma sources and the energy from these sources. Because the UF_6 would originate from reprocessed uranium, levels of radiological source terms (gamma and neutron), and other radiation sources, impurities, etc., would need assessment to establish shielding requirements for transportation of larger quantities of reprocessed UF_6 up to 20 weight percent U-235. For criticality evaluations of packages containing reprocessed UF_6, estimates of radionuclide compositions, and their distribution throughout the package (e.g., inhomogeneous distribution of impurities, possible voids, and density of contents), would need to be specified to evaluate criticality safety for larger quantities of UF_6, such as that proposed for the DAHER 30B-20. Limited criticality experiments for HALEU assays that consider realistic transportation package configurations with contents to 20 weight percent U-235 exist and would prompt the need for an applicant to either use conservative design assumptions, or obtain additional criticality data in order to reduce uncertainty with existing benchmark data used to validate criticality evaluations.

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