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Technical Information Needs and Regulatory Considerations for Front-End Transportation Activities of HALEU Fuel Feed Material

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Prepared for the U.S. Nuclear Regulatory Commission
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Pacific Northwest National Laboratory
Richland, Washington 99354

Summary

This report fulfills task order number *31310022F0033 Task 1 Assessment of technical information needs and considerations for front-end transportation activities for HALEU fuel feed material*. This assessment primarily focuses on the outlook and packaging requirements for transporting high-assay low-enriched uranium (HALEU) fuel feed material. Section 3.0 describes the regulatory framework. Section 4.0 discusses the specific issues associated with reviewing HALEU feed material packages. The rest of the report discusses the need for new packages and Department of Energy progress on HALEU production. The evaluation showed that the 5 percent enrichment barrier between low-enriched uranium and HALEU is a significant challenge to transportation packaging operations because previous standards didn't allow for this without additional analyses and newer standards have not been used by the NRC yet. The NRC should expect and be ready for new packaging and facilities reviews to support this change. The reviews may be more in depth than previous and although small amounts of HALEU feedstock can be easily transported, moving large amounts will require new packages. However, there are licensed designs and concepts that can be used to transport small amounts of HALEU feedstock. No significant technical issues were identified for transporting material through the entirety of the HALEU range up to 20 percent enriched.

Acronyms and Abbreviations

A_2	maximum activity level of normal form radioactive material
ANSI	American National Standards Institute
ARDP	Advanced Reactor Demonstration Project
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	Boiler and Pressure Vessel Code
CFR	Code of Federal Regulations
CoC	certificate of compliance
CSI	criticality safety index
DOE	Department of Energy
DOT	Department of Transportation
EBR-II	Experimental Breeder Reactor-II
HAC	hypothetical accident conditions
HALEU	high-assay low-enriched uranium
HEU	highly enriched uranium
IAEA	International Atomic Energy Agency
k_{eff}	reactivity coefficient, the effective neutron multiplication factor
LEU	low-enriched uranium
LEU+	low-enriched uranium enriched between 5 and 10 percent by weight of U-235
LWR	light-water reactor
MARVEL	Microreactor Applications Research Validation and Evaluation
MOU	memorandum of understanding
NAC	Nuclear Assurance Company
NCT	normal conditions of transport
NNSA	National Nuclear Security Administration
NRC	U.S. Nuclear Regulatory Commission
NUREG	NRC technical report
OV	oxide vessel
PNNL	Pacific Northwest National Laboratory
QA	quality assurance
QAP	quality assurance plan
RG	regulatory guide
SAR	safety analysis report
SARP	safety analysis report for packaging
SNF	spent nuclear fuel

SNM	special nuclear material
SSR	specific safety requirement
TRISO	tristructural isotropic
UF_4	uranium tetrafluoride
UF_6	uranium hexafluoride
UO_2	uranium dioxide
USEC	United States Enrichment Corporation (now Centrus Energy)
VPC	valve protective cover
ZIRCEX	Hybrid Zirconium Extraction

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1.0 Introduction

This report is in fulfillment of task order 31310022F0033, *Task 1 Assessment of technical information needs and considerations for front-end transportation activities for HALEU fuel feed material.*

The U.S. Nuclear Regulatory Commission (NRC) has tasked Pacific Northwest National Laboratory (PNNL) with evaluating potential challenges associated with the safe production, storage, and transportation of advanced non-light-water reactor fuel forms and fuel materials, and determining the information needs in the context of existing regulatory frameworks and guidance. Many of these reactors are designed to use high-assay low-enriched uranium (HALEU). This is fuel enriched with U-235 greater than 5 percent but not greater than 20 percent. Current light-water reactors (LWRs) with low-enriched uranium (LEU) have U-235 enrichments up to 5 percent. Because of this 5 percent limit, the LWR fuel cycle is designed around production and shipment of LEU fuel. This includes enrichment facilities, fuel fabrication facilities, and transportation packages used to move material. Many of these systems are designed with a regulatory or safety basis framework that does not consider enrichments above 5 percent. This could be due to specific regulatory exemptions and restrictions or simple lack of need for HALEU-compatible systems up to this point.

Specifically, this report addresses the safety and regulatory considerations for the transportation of the feed material that will be used to make HALEU fuel. Primarily, this material is uranium tetrafluoride (UF_4) and uranium hexafluoride (UF_6), which would be shipped from an enrichment or downblending facility to a fuel fabrication facility. This report evaluates the current state of HALEU production and transportation, along with the associated regulatory framework. Any gaps or needs for additional guidance and information are identified in the conclusions and recommendations section.

2.0 HALEU Background

All United States commercial LWRs and supporting fueling infrastructure currently utilize LEU – uranium enriched with the uranium isotope U-235 up to 20 weight percent – as a material feed fuel source. Though LEU fuel used in these cases is limited to a maximum of 5 percent enrichment according to 10 CFR 50.68, HALEU is defined as uranium enriched to between 5 and 20 percent with U-235 (42 USC 16281).

Several recently proposed advanced reactor designs feature the use of HALEU fuel. This decision is likely motivated by the operational safety and efficiency potential offered by a reduced core size as well as the significant extended duration between refueling outages that could be provided. Additionally, several fuel vendors are proposing accident tolerant fuels for current LWRs that would utilize HALEU fuel to leverage enhanced safety and extended operations characteristics as well. Two general methods of producing HALEU currently exist. The first method is to enrich virgin, naturally occurring uranium harvested from ore, which contains less than 1 percent U-235, to achieve the higher 5 to 20 percent U-235 enrichment range needed. The exclusive commercial enrichment method used with virgin material within the United States leverages gas centrifuge technology (NRC 2020d) and is currently only used to achieve up to 5.5 percent at one location (Freels 2021). As of 2023, the NRC is actively engaged with American Centrifuge Operating, LLC, a wholly owned indirect subsidiary of Centrus Energy, to authorize production of HALEU at a demonstration facility known as the American Centrifuge Lead Cascade Facility located on a Department of Energy (DOE) reservation in Piketon, Ohio.[1] Laser separation technology is another option to consider, but it is still under development and has yet to be demonstrated to be able to support commercial scale production (NRC 2020d).

The second method of producing HALEU is by using an existing stockpile of highly enriched uranium (HEU), which has enrichments greater than 20 percent, and downblending it with natural uranium or other LEU sources. However, there is no readily available or identifiable supply chain of HEU that will meet potential demand projections. Additionally, the current non-defense-related HEU supply is predominantly dedicated and overcommitted to research reactor fuel production to support advanced test reactor development, tritium production, and medical isotope production, among other things. To support near-term advanced reactor development, DOE and its national laboratories are exploring producing HALEU fuel feed material via recovery of uranium from spent nuclear fuel (SNF) from DOE research reactors using two recycling methods.

The first recycling method uses electrochemical processing to recover and polish source material feed. The second method applies the Hybrid Zirconium Extraction (ZIRCEX) process. Either could be used effectively to recover HEU. This feed material could then be down-blended to make HALEU fuel feed material. Electrochemical processing tends to leave more residual radioisotopes than the ZIRCEX process. Table 1 represents the expected average HALEU fuel feed material composition of recovered and polished Experimental Breeder Reactor-II (EBR-II) SNF after appropriate downblending (Vaden 2021). The ZIRCEX processes will likely have lower amounts of residual radioisotopes. As discussed in Section 4.0, specific packaging challenges stem from the greater number of residual radioisotopes that exist after electrochemical processing to recover and polish source material feed.

[1] Centrus Energy Corp./American Centrifuge Operating, LLC (formerly USEC Inc.) Gas Centrifuge Enrichment Facility Licensing, https://www.nrc.gov/materials/fuel-cycle-fac/usecfacility.html.

Table 1. Average Material Composition of EBR-II HALEU (Vaden 2021)

Analyte	Units	Average
Total U	wt%	99.95
U-232	ppbU	0.66
U-233	ppbU	49.12
U-234	iso% U	0.17
U-235	iso% U	19.39
U-236	iso% U	0.58
U-237	pptU	0.06
U-238	iso% U	79.86
Zr	ppm	101.45
Si	ppm	77.6
Y	ppm	6.05
Fe	ppm	133.13
Cr	ppm	28.15
Ni	ppm	43.14
Mo	ppm	40.21
Mn	ppm	79.47
Ru	ppm	77.23
Cd	ppm	12.17
Al	ppm	101.24
Tc	ppm	75
Li	ppm	15.6
K	wt%	0.05
Na	ppm	82.29
Ba	ppm	5
Sr	ppm	5
Sr-90	ppb	15.77
Nd	ppm	95.92
Sm	ppm	56.82
Tc-99	ppm	0.15
Cs-135	ppm	2.67
Mn-54	ppt	3.04
Co-60	ppt	27.81
Sb-125	ppt	102.51
Cs-134	ppt	24.99
Cs-137	ppb	8
Ce-144	ppt	67.11
Eu-154	ppb	0.22
Eu-155	ppb	0.22
Am-241	ppb	61.23
Np-237	ppm	17.11
Pu-239	ppm	83.57
Pu-240	ppm	2.24

3.0 Regulatory Framework

The front end of the fuel cycle is regulated through multiple agencies in the United States. These include the Department of Transportation (DOT), NRC, U.S. Coast Guard, and the relevant state agencies. Internationally, the International Atomic Energy Agency (IAEA) has established transportation regulations, and these standards inform the NRC framework. There are also packages certified under IAEA standards used in the United States in certain circumstances. Therefore, the IAEA framework is discussed in this report.

Due to the overlap in the statutory authorities of NRC and DOT, the two agencies signed a memorandum of understanding (MOU) in 1979 regarding regulation of the transport of radioactive material. The principal objective of the MOU was to avoid conflicting and duplicative regulations and to clearly delineate the areas in which each agency establishes regulations (PHMSA 2008). The NRC regulations relevant to transport are 10 CFR Part 71 and 10 CFR Part 73. Part 71 establishes safety regulations for packaging and transportation and part 73 establishes security regulations for transportation of radioactive materials.

3.1.1.1 10 CFR Part 71

In 10 CFR Part 71, NRC has promulgated regulations for the packaging and transportation of radioactive material that align with its safety mission of protecting the public and the environment. NRC has adopted by reference (10 CFR 71.5) portions of the DOT regulations, enabling NRC to inspect its licensees for compliance with DOT regulations applicable to shipper/licensees and to take enforcement actions on violations. Transportation packages are approved to travel over road, rail, waterway, and air.

NRC package approval standards include general standards applicable to all packages and requirements related to the normal conditions of transport (NCT) and to hypothetical accident conditions (HAC). The package types reviewed by the NRC are Type AF, B, and BF. Type A packages and IP packages are reviewed by the DOT. Special attention is given in 10 CFR Part 71 to Type B packages containing more than 10^5 A_2 (a measure of hazard) and plutonium air shipments. These types are defined in 10 CFR Part 71.

Regardless of the package type, the design focus of any transportation package is to meet regulations for dose, criticality, and release. NCT (10 CFR 71.71) and HAC (10 CFR 71.73) conditions embody these concepts by requiring the package to meet various requirements via drop tests, thermal tests, and water immersion, among others. Lifting requirements and package tie-down requirements must also be met. Design specifics not contained in the regulations themselves are often found in the standard review plan issued by the NRC in technical report NUREG-2216, which cites other NUREGs and regulatory guides (RGs).

If the NRC determines that a transportation package meets 10 CFR Part 71 requirements based on submission by an applicant via a safety analysis report for packaging (SARP), a certificate of compliance (CoC) is issued. The CoC is valid for 5 years and can be renewed. The CoC describes the package and its contents and incorporates license drawings, package operations, and the acceptance and maintenance program from the safety analysis report.

3.1.1.2 10 CFR Part 73

In 10 CFR Part 73, NRC has promulgated regulations for the physical protection of plants and materials. Included in these regulations are requirements for the physical protection of special nuclear material (SNM) including irradiated reactor fuel in transit. The objective of the physical protection system for shipments is to minimize the potential for theft, diversion, or radiological sabotage and to facilitate the location and recovery of shipments that may come under control of unauthorized individuals.

NUREG-0561 (NRC 2013) provides detailed guidance for the physical protection of irradiated reactor fuel in transit, including route approval procedures, preplanning and coordination of shipments, advance notification of shipments, communications, arrangements with local law enforcement agencies, armed escorts, shipment logs, procedures training, and control of information. Further, NUREG-0561 (NRC 2013) presents requirements by transport mode (e.g., road, rail, and vessel), and discusses background investigations for personnel with unescorted access to SNF in transit.

3.1.1.3 *Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material*, NUREG-2216 (NRC 2020c)

NRC regulations in 10 CFR Part 71 require that radioactive material packaging be able to remove heat generated by radioactive decay, shield the public from radioactive particle emissions, ensure nuclear subcriticality control, and mitigate the dispersion of radioactive material. Challenges associated with transporting HALEU feed material and HALEU primarily revolve around the last two requirements. Design and licensing of packages within the United States can involve the application of a number of consensus standards and guidance, such as applicable American Society of Mechanical Engineers (ASME), American National Standards Institute (ANSI), and American Society for Testing and Materials (ASTM) national standards, and guidance from NRC regulatory guides. These flow down requirements and dictate aspects of mechanical design, material selection, fabrication (including welding), examination, testing, SARP preparation, and certification.

NRC RG 7.9 provides a suggested format for a SARP and documenting the package evaluations to establish the safety basis (NRC 2005b). NRC has also issued a standard review plan for transportation packages in NUREG-2216 (NRC 2020a). The guidance is separated by technical discipline and submittal format. The main application areas to review are structural, thermal, containment, shielding, criticality, materials, operating procedures, acceptance testing and maintenance, and quality assurance (QA). This section discusses the standard review plan chapters and their general requirements as they relate to all radioactive materials packages. Section 4.0 further describes the specific aspects related to HALEU feedstock packages and makes recommendations for reviews of these packages.

Structural

Shielding, criticality, and thermal disciplines each revolve tightly around the structural evaluation discipline for their safety basis. Approval standards in 10 CFR 71.41 state that to demonstrate compliance, the effects of tests specified in 10 CFR 71.71 (NCT) and 10 CFR 71.73 (HAC) must be evaluated. The component safety groups can be further divided into containment, criticality, and other safety components, as defined below.

Containment components are defined as all the components required for retaining the nuclear and possibly radioactive contents. The function of all the containment vessel and closure components is to maintain the containment boundary so that all NCT and HAC containment requirements are met. This component safety group includes any closure lids, seals, port components, and bolts. Type B, Category I packaging design criteria dictates that the structural evaluations and stress limits need to comply with the ASME Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, Subsection NB per NUREG/CR-3854 (Fischer and Lai 1985). Stress limits differ between NCT and HAC such that meeting NCT structural performance requirements tends to be more challenging and restrictive than meeting HAC structural performance requirements.

For Type B, Category 1 packaging, analytical methods should be used to demonstrate that the containment vessel meets the vessel design criteria presented in Subsection NB, Article NB-3000 as amended by RG 7.6 (NRC 1978). Also, in the case of closure bolts, due to their importance, applicants/developers are directed to use acceptance criteria provided in NUREG/CR-6007 (Mok et al. 1992). Limits for release of contents are specified in 10 CFR 71.51 for both NCT and HAC evaluations. Demonstration of compliance with the specified limits must be in accordance with the methods laid out in ANSI N14.5. Additionally, in accordance with 10 CFR 71.61, a Type B package containing more than 10^5 A_2 must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than 1 hour without collapse, buckling, or inleakage of water.

Criticality components are defined as all components required in controlling nuclear criticality during transport of fissile material in the package. This component safety group includes neutron absorbers and related structures required to retain the relative position of the fissile materials and/or neutron absorbers. Type B, Category I packaging design criteria dictates that the structural evaluations and stress limits need to comply with ASME BPVC, Section III, Division 1, Subsection NG (ASME 2019a) per NUREG/CR-3854 (Fischer and Lai 1985).

Other safety components are defined as all other safety-related packaging components. This includes both gamma and neutron shielding components (if required); secondary seals, bolts, and closures; impact limiters; and lifting lugs and tie-down devices. Type B, Category I packaging design criteria dictates that the structural evaluations and stress limits need to comply with ASME BPVC, Section VIII, Division 1 (ASME 2019c), or Section III, Subsection NF (ASME 2019b) per NUREG/CR-3854 (Fischer and Lai 1985).

ASME BPVC, Section III, Division 1, Subsection NB (ASME 2015) contains the requirements for materials, design, fabrication, examination, and testing of Class 1 reactor components to form acceptable design criteria for shipping containment vessels. A critical consideration is that the choice of materials is limited, and the temperature use limits are 700°F for carbon and low alloy (ferritic) steels and 800°F for austenitic and high alloy steels.

Thermal

Similar to the structural discipline, the thermal requirements of 10 CFR Part 71 and 49 CFR Parts 100-185 must be met before a package can be certified for the transportation of nuclear material. Packaging must be capable of withstanding intense thermal environments while preventing reconfiguration or release of the contents and nuclear subcriticality. Achieving this capability requires the use of construction materials that enable the package to withstand serious thermal insult. Accordingly, all Type B, Category 1 safety significant components of packaging must be designed such that all containment items withstand the highest expected temperature under HAC after the entire structural impact loading sequence per NRC RG 7.9 (NRC 2005b). After the HAC structural and thermal loading sequences of the packaging definition, a maximum leak rate of an A_2 per week is permissible under HAC as specified in 10 CFR 71.51(a)(2).

Containment

The design of a Type B, Category 1 packaging definition typically starts with the containment system. The containment system is defined as an assemblage of all the components required to retain the nuclear and possibly radioactive contents. In general, this includes items such as the containment vessel, possible seals and port components, and closure bolts. The containment boundary is an assemblage of all the components required to retain the contents and is in direct communication with the internal cavity of the containment vessel. The structural design criteria for certain package component safety groups are based on the ASME BPVC as discussed in the Structural section.

Containment design requirements for Type B, Category 1 packaging designs that are used to transport radioactive materials must be developed to ensure that any release of radioisotopes during postulated NCT or HAC events falls within the specified regulatory limits. Primary regulatory documents that provide the general requirements for containment are 49 CFR Parts 100-185 and 10 CFR Part 71, and condition-specific requirements are listed in 10 CFR 71.51 and 10 CFR 71.71 for NCT and in 10 CFR 71.51 and 10 CFR 71.73 for HAC. Containment requirements are specified in 10 CFR 71.43, “General standards for all packages.” The phrase “no loss or dispersal of radioactive contents” is clarified in 10 CFR 71.51, “Additional requirements for Type B packages.”

Although shipping packages are designed to contain radioactivity and to maintain their structural integrity under the most severe reasonably anticipated conditions, leak testing is necessary to ensure that the packages are manufactured and assembled correctly and that no unacceptable leak paths have developed from subsequent use (Anderson et al. 1996). 10 CFR 71.51 specifies the limits for maximum permissible rate of release of contents for both NCT and HAC as 10^{-6} A_2 per hour and one A_2 in one week, respectively. Additionally, NCT testing is to be conducted at the most unfavorable conditions of external temperature (between −29°C [−20°F] and +38°C [+100°F]) and pressure (between 25 kPa [3.5 psi] absolute and 140 kPa [20 psi] absolute).

ANSI N14.5-2014 is critical to compliance because it provides a bridge from the regulatory release rate requirement to a measurable allowable leakage rate. ANSI N14.5-2014 also provides a list and descriptions of several accepted leakage rate test methodologies (ANSI 2014). Frequently, designers/developers will select a leaktight definition (ANSI N14.5-2014 defines a leakage rate of 10^-7 ref-cm^3/s or less as leaktight) as a less-complex means of determining acceptance. If this is done, then establishment of a content-dependent leakage rate determination is not required.

Leakage rate testing is required prior to first use, periodically (annually for standard definition Type B packaging) during package life and following maintenance or repair activities. Leakage rate tests may also be required during design and associated verification testing, fabrication, and preshipment. National Nuclear Security Administration (NNSA) Safety Guide (SG)-100 (NNSA 2005) describes and gives examples of leak testing methods and supported details to consider while applying ANSI N14.5-2014 (ANSI 2014).

Shielding

Packaging radiation shielding design is concerned with establishing that the regulatory radiation dose rate limits on the exterior of the packaging are not exceeded. The same calculations that produce radiation source term evaluations for shielding analyses are also typically used to determine the heat sources used in the thermal analyses. Shielding performance requirements are listed in 10 CFR 71.47 and 10 CFR 71.51.

10 CFR 71.47 specifies dose rate limits for packaging anticipated to be used for transport. Under NCT, the radiation dose rate does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package, as specified by 10 CFR 71.47, 49 CFR 173.441, and IAEA Specific Safety Requirement (SSR)-6. The maximum dose rate at 2 meters from any external surface position of the vehicle under NCT must also not exceed 0.1 mSv/h (10 mrem/h).

The NCT dose rate limits apply to a shipping package without regard for the method of shipment. However, most packages for fuel feed material are shipped as “exclusive use” as defined in 10 CFR 71.4. If the package is shipped as exclusive use, the NRC limits can be relaxed to account for the material and geometric shielding properties of the conveyance vehicle. A maximum package external dose of 10 mSv/h (1,000 mrem/h) for NCT is allowed in a closed vehicle if the 2 mSv/h (200 mrem/h) limit is met on the external surface of the vehicle. The details of exclusive use limits are given in 10 CFR 71.47(a) and (b).

The maximum dose rate at 1 meter from any external surface position of the package under HAC must not exceed 10 mSv/h (1,000 mrem/h), as specified in 10 CFR 71.51(a)(2). For HAC, the dose rate of the external package surface is assumed to be that defined by the post-accident configuration.

Criticality

Subcriticality design and performance requirements are described in 10 CFR 71.55 through 10 CFR 71.61. Fissile material is defined in 10 CFR Part 71 and SSR-6 as material that contains the nuclides Pu-239, Pu-241, U-233, and U-235. Criticality safety is the practice of ensuring that adequate protection is provided against an accidental self-sustaining or divergent fission chain reaction. The criticality state of a system (subcritical, critical, or supercritical) is often discussed in terms of an effective neutron multiplication factor, K effective (k_{eff}), which is defined as the ratio of the neutron production rate to the neutron loss rate in the system. For the system to remain subcritical, k_{eff} must be less than unity.

The conditions prescribed by the regulations require computational evaluations that incorporate statistical techniques to model neutron transport and predict k_{eff} for the system. Calculational biases and uncertainties are also part of this determination. The margin of subcriticality allowed for a package configuration must include the effect of these biases and uncertainties, together with design uncertainties, and an additional subcritical margin that would provide subcriticality for all credible transport scenarios even in the absence of all uncertainties.

Generally, criticality control is provided by geometric spacing and strategic poisoning. As such, these factors are always considered to be safety significant and must be demonstrated to remain intact during NCT as well as HAC evaluation scenarios. Another factor in criticality analysis is the potential for water inleakage to provide a moderator to the package. To disposition this effect, flooded conditions must be analyzed or the package containment must be proven to exclude moderator entry.

Restriction of fissile mass or use of a favorable geometry to provide enhanced neutron leakage from the package definition as a means of controlling the neutron balance are not completely feasible for establishing a packaging convention capable of supporting commercial-scale transport needs. Instead, strategic incorporation of neutron poison materials and moderators is the primary means of controlling neutron balance. Neutron poisons added to the package definition require special attention because their presence must be certain under all conditions and because their incorporation may change the mechanical and/or thermal properties of host materials. Additionally, the geometry of heterogeneous fissile material (e.g., tristructural isotropic fuel compacts), the design and placement of absorber materials, and the separation between fissile materials are all important to the criticality evaluation.

Whatever the control mechanism, an adequate margin of subcriticality must be demonstrated for both the single package in isolation and for arrays of packages that might be required for economic commercial-scale transport. Undamaged (normal transport conditions – NCT) and damaged (subsequent to accident conditions – HAC) packages must be considered using the credible fissile material configuration and the moderator and reflector conditions that provide the maximum effective neutron multiplication factor, k_{eff}.

Criticality safety evaluations must be performed in establishing the safety basis of the packaging definition to demonstrate that the package will remain in a subcritical condition under NCT (see 10 CFR 71.71) and HAC (see 10 CFR 71.73). Domestic and international regulations require that a single, water-flooded package be adequately subcritical in both the undamaged and damaged condition. 10 CFR 71.55 describes the specific domestic requirements that must be met for a single package to be certified to carry fissile material.

The undamaged package is considered to be the physical condition of the package under NCT; the damaged package refers to the physical condition of the package following its exposure to the tests for HAC. All internal voided volumes of the package, including the containment vessel, must be assumed to be filled with optimum water moderation for this evaluation unless moderator exclusion is successfully established as the foundation of the safety basis. The fissile material contents used in the evaluation must also be in the most reactive credible condition consistent with NCT and HAC. All forms of hydrogenous moderation are intended for consideration in determining the optimum moderation (i.e., moderation conditions for highest k_{eff} value).

Regulations state that the criticality safety evaluation must include an analysis to determine whether the portion of the package defined as the containment system, if closely reflected by water, would have a greater reactivity (higher k_{eff}) than the packaging in combination with water. If the package and containment system are not the same (e.g., 30B or 30C canister and UX-30 overpack), then two analyses must be done for the package in its undamaged condition—one with a water-flooded and water-reflected containment system separate from the package and one with a water-flooded and water-reflected package. The results from these two system analyses should be compared to select the one with the highest k_{eff} as the limiting case for the certification process. In short, the packaging review must evaluate all possible moderation and reflection configurations as part of establishing and verifying the safety basis.

A requirement included in the domestic 10 CFR 71.55(f) and international regulations applies to packages containing fissile material that will be shipped by air. This requirement is provided to preclude a rapid approach to criticality that may arise from potential geometrical changes in a single package (thus, water inleakage does not need to be considered) after the tests (developed to be consistent with HAC that might arise from air transport) prescribed in 10 CFR 71.55(f).

Materials

The materials evaluation is interrelated with many other safety analyses, including thermal, structural, and containment. This evaluation interacts with these other chapters in both directions because it determines whether the materials used in package construction will be adequate for performing their safety function and evaluates the materials in light of the requirements for structural, thermal, and shielding performance. Additionally, there must be no adverse interactions caused by material type or construction. The materials are evaluated for NCT and HAC conditions. In general, high-temperature performance is a critical factor in HAC. In NCT, this is not as impactful unless the contents generate a lot of decay heat. The materials evaluation also includes a review of weld design and the inspection of those welds. These feed into the containment evaluation along with seal and bolt materials.

Operating Procedures

The operating procedures evaluation covers the loading and unloading procedures for packaging. This ensures that safety is maintained during loading and unloading and that any inspections and tests are sufficient to ensure that the packaging will perform as required. An essential focus of this review is making sure that the closure and containment systems will perform as necessary and required by the other technical disciplines. This might include bolt torques, valve tests and set points, seal inspections, and leak tightness inspections.

Acceptance Tests and Maintenance

The acceptance tests and the maintenance procedure review are designed to evaluate the procedures used to determine whether package materials and construction are adequate to perform their safety and design functions. Acceptance tests are required to verify all the performance requirements, including structural, thermal, and criticality. This verifies that the fabrication of the package was appropriate. For the maintenance evaluation, the applicant must ensure that long-term performance of the package materials and construction is managed appropriately. This may be through inspections and/or scheduled maintenance and replacement of components.

Quality Assurance

Quality assurance throughout the design process is as important to the successful certification of a package as the package’s ability to successfully complete the regulatory testing and verification of the safety basis (simulated or physical). DOE and NRC certifying bodies require applicants for packaging certification and shipment to adhere to 10 CFR Part 71, Subpart H requirements. QA programs based on implementation of a full-featured Nuclear Quality Assurance-1 program (ASME 2019) are acceptable as well.

A description requirement to partially address 10 CFR 71.31 is to identify established codes and standards proposed for use in the package design, materials of fabrication, fabrication, assembly, testing, maintenance, and use. In the absence of codes, standards, and applicable code cases, the basis and rationale used to formulate the QA program need to be described in adequate detail and justified to the regulatory faction (10 CFR Part 71).

A quality assurance plan (QAP) developed for the program should address all 18 elements required and identify the procedures that will be used to achieve the applicable development quality requirements for the packaging definition. The QAP should be developed, submitted to the certifying regulatory body for approval, and available for direct application prior to the initiation of the design and development effort.

QA specialists should have experience with packaging regulations, DOE orders, and national and international standards relating to QA in addition to 10 CFR Part 71, Subpart H requirements. The applicant's QA program should detail their approach to the control of purchased items and services in order to fulfill the requirements of 10 CFR 71.115, "Control of purchased material, equipment, and services," and 71.109, "Procurement document control." Vendors should be carefully selected based on their capability to comply with applicable sections of 10 CFR Part 71, Subpart H, their facility and QA program, and their previous records and performance. Also, all activities related to fabrication should be documented in a SARP and conducted under a certifying regulatory body's approved QA program.

The NRC published RG 7.10 Revision 3 (NRC 2015) to provide individuals subject to the QA requirements of 10 CFR Part 71, Subpart H guidance on developing QA programs for implementation with respect to the transport of radioactive materials in Type B and fissile material packagings, such as those potentially required for HALEU feed material and HALEU transport. RG 7.10 establishes a graded quality safety category system that delineates packaging definition items between 1) critical to safe operation, 2) has a major influence on safety, 3) only has a minor influence on safety.

Radiation shielding evaluation aspects, all physical testing initiatives, leakage rate testing, and instrument calibration also need to be performed in accordance with a written and accepted QA program and QAP that conform with the applicable requirements of 10 CFR Part 71, Subpart H, and other relevant codes and standards. Additionally, measures must be established to ensure that test results are documented, evaluated, and maintained as QA records to meet the requirements of 10 CFR 71.123.

3.1.1.4 Additional NRC Regulatory Guides and Standard Review Plans

NRC regulatory guides provide direction to licensees and applicants on implementing specific parts of the NRC regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. Regulatory guides specific to transportation include:

- RG 7.4, Revision 1, *Leakage Tests on Packages for Shipment of Radioactive Materials* (NRC 2020b).
- RG 7.6, Revision 1, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels* (NRC 1978).
- RG 7.7, Revision 1, *Administrative Guide for Verifying Compliance with Packaging Requirements for Shipments of Radioactive Materials* (NRC 2012).
- RG 7.8, Revision 1, *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material* (NRC 1989).
- RG 7.9, Revision 2, *Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material* (NRC 2005b).

- RG 7.10, Revision 3, *Establishing Quality Assurance Programs for Packaging Used in Transport of Radioactive Material* (NRC 2015).
- RG 7.11, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)* (NRC 1991a).
- RG 7.12, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)* (NRC 1991b).

In addition, the IAEA publishes safety standards and guidance in the form of SSRs.[1]

3.1.1.5 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"

The requirements in 10 CFR Part 70 establish the licensing criteria to "establish procedures and criteria for the issuance of licenses to receive title to, own, acquire, deliver, receive, possess, use, and transfer SNM; and establish and provide for the terms and conditions upon which the Commission will issue such licenses." In practical terms, these are the regulations that govern facilities for the production and manufacture of fuel and fuel materials. One critical aspect of Part 70 is the categorization of fuel facilities. This category is based on the amount of SNM at the licensed facility. Since a facility producing HALEU would be Category 2 and all current, commercial U.S. nuclear fuel fabrication facilities are Category 3, a Category 2 fuel fabrication facility would need to be licensed. Categorization determines the extent of the safety and security requirements.

[1] One of these documents –SSR-6, *Regulations for the Safe Transport of Radioactive Material* (IAEA 2018) – is the governing document for transportation and transportation packaging. Although the U.S. transportation and transportation packaging are regulated by 10 CFR Part 71, SSR-6 is still important to review because 10 CFR Part 71 is designed to be compatible with SSR-6 and there are packages that may be certified under SSR-6 standards and used for international shipments to the United States.

Table 2. 10 CFR Part 70 Material Categories

Categories Defined in 10 CFR Part 70	Definition from 10 CFR Part 70
Category 1	Formula quantity: 5,000 grams using the formula, grams = (grams U-235 (HEU >20%)) + 2.5 (grams U-233 + grams plutonium)
Category 2	SNM of moderate strategic significance: (1) Less than a formula quantity of strategic special nuclear material but more than 1,000 grams of U-235 (contained in uranium enriched to 20% or more in the U–235 isotope) or more than 500 grams of U-233 or plutonium, or in a combined quantity of more than 1,000 grams when computed by the equation, grams = (grams contained U–235) + 2 (grams U–233 + grams plutonium); or (2) 10,000 grams or more of uranium-235 (contained in uranium enriched to 10% or more but less than 20% in the U–235 isotope).
Category 3	SNM of low strategic significance means: (1) Less than an amount of special nuclear material of moderate strategic significance as defined in paragraph (1) of the definition of strategic nuclear material of moderate strategic significance in this section, but more than 15 grams of uranium-235 (contained in uranium enriched to 20 percent or more in U–235 isotope) or 15 grams of uranium-233 or 15 grams of plutonium or the combination of 15 grams when computed by the equation, grams = (grams contained U–235) + (grams plutonium) + (grams U–233); or (2) Less than 10,000 grams but more than 1,000 grams of uranium-235 (contained in uranium enriched to 10 percent or more but less than 20 percent in the U–235 isotope); or (3) 10,000 grams or more of uranium-235 (contained in uranium enriched above natural but less than 10 percent in the U–235 isotope).

4.0 Transportation Review for HALEU Packages

This section discusses aspects of transportation packaging that are specific to UF_6, UF_4, and the HALEU versions of these materials. Each technical area is evaluated and recommendations about possible guidance updates and technical work are made. The general regulations associated with transporting nuclear material do not have HALEU-specific aspects; these include the DOT, U.S. Coast Guard, and state regulations. For example, although a HALEU package may be larger than standard UF_6 cylinders, the overweight transport considerations would be the same as those for any other package. For this reason, focus is on the NRC regulations and guidance. NUREG-2216 (NRC 2020c) incorporates interim staff guidance and guidance from other sources. It also references several NRC information notices (IN-92-58, IN-97-24, IN-97-20, and IN-16-06) (NRC 2016) and Bulletin 94-02 (NRC 1992, 1994, 1997b, 1997a). These documents provide additional details on safety issues relevant to the transport of UF_6 packages. This section is organized by technical discipline, starting with a general design overview.

4.1 General Package Design Considerations

Uranium hexafluoride packages typically consist of an inner steel cylinder that acts as a containment vessel and an outer protective overpack. The protective overpack is typically required for the shipment of enriched UF_6 packages. As discussed in NUREG-2216, Appendix A.6, the inner cylinder is carbon steel with rounded ends and a protective skirt. On one end of the cylinder is a valve for filling and emptying the cylinder; on the other end is a removable plug. The most commonly used commercial cylinders are approximately 0.76 m (30 in.) in diameter and 2.1 m (81 in.) in length with a capacity of about 2,300 kilograms (2.5 tons) of UF_6 (NRC 2020c).

As discussed in NUREG-2216, Appendix A.6, the protective overpack is generally a double-shell, stainless-steel cylinder with cushioning pads on the inner cavity. An energy-absorbing, insulating foam fills the space between the inner and outer shells. The overpack can be separated into two halves to enable easy access to the inner cylinder. Overpacks for the 30 in. cylinders mentioned above are approximately 0.1016 m (4 in.) thick (NRC 2020c). United States Enrichment Corporation (now Centrus Energy) (USEC)-651 (USEC 2017) contains information regarding overpacks.

The design and authorized contents of UF_6 cylinders are defined in ANSI N14.1, *Nuclear Materials—Uranium Hexafluoride—Packagings for Transport*. In addition, in 49 CFR 173.420(a)(2)(i), the DOT requires that the UF_6 packages be "designed, fabricated, inspected, tested and marked in accordance with—(i) American National Standard N14.1 in effect at the time the packaging was manufactured." Typically, enriched UF_6 is shipped from enrichment facilities to fuel fabrication facilities in 30B cylinders; the previous (2019) revision of ANSI N14.1 specified a maximum enrichment of 5 percent for 30B and 30C[1] cylinders. ANSI N14.1-2023 modified this restriction to allow higher enrichments with additional criticality evaluation.

For UF_6 packages, the primary safety function of the cylinder is to provide containment and moderation control for criticality purposes. Moderation control is required for all commercially

[1] The ANSI N14.1 Standard 30C cylinder is essentially a 30B cylinder equipped with a valve protective cover (VPC) that bolts over and protects the cylinder valve during transport. The VPC is a special design feature that provides additional assurance against the inleakage of water to the containment system and is an enclosure that retains any leakage.

used cylinders for fissile UF_6 and must be maintained under NCT and HAC. To assure subcriticality by moderation control, the mass of the contents must be at least 99.5 percent UF_6 (NRC 2020c).

As discussed in NUREG-2216, Appendix A.6, the cylinder is defined as the containment boundary for the UF_6. Unirradiated uranium enriched to less than 5 percent is a Type A quantity. Recycled uranium can be a Type B quantity due to the presence of uranium-232, uranium-234, uranium-236, and various radioactive impurities, especially transuranic radionuclides.

For UF_6 packages, shielding requirements are generally not significant because of the low radioactivity and self-shielding of UF_6 (NRC 2020c). If the contents are recycled uranium, a shielding evaluation is required to show that the package will meet the dose rate limits in 10 CFR 71.47, "External radiation standards for all packages," and 10 CFR 71.51, "Additional requirements for Type B packages, during NCT and HAC," respectively (NRC 2020c).

The overpack provides thermal protection to prevent overheating of the UF_6, which can cause hydraulic failure of the cylinder. The overpack also provides impact protection for the cylinder and the valve (NRC 2020c).

For UF_6 packages, NUREG-2216 (NRC 2020c) lists the following safety features:

- The steel cylinder precludes inleakage of water and provides containment under normal conditions of transport and HACs.
- The cylinder skirt provides some protection for the valve during handling operations, NCT, and HAC.
- The overpack provides structural and thermal protection for the cylinder and its valve under HAC.

Typical areas of review involving overpack drawings are (NRC 2020c):

- Overpack shell
  - Materials of construction
  - Dimensions and tolerances
  - Vents for pressure relief of foam combustion products
- Foam specifications
  - Type
  - Density
  - Compressive strength
  - Fire retardant characteristics
  - Limit on free chlorides
- Closure devices
  - Torque
  - Valve protection device

4.2 Structural

As with any transportation package, the structural review of such a package supports conclusions made on dose, critically, and release. According to the regulations, 10 CFR 71.55(g) discuses several criteria that can be excepted from a UF_6 package with respect to water inleakage for traditional UF_6 material with a maximum 5 percent enrichment of U-235. From a structural point of view, 10 CFR 71.55(g)(1) is of importance:

> *Following the tests specified in § 71.73 ("Hypothetical accident conditions"), there is no physical contact between the valve body and any other component of the packaging, other than at its original point of attachment, and the valve remains leak tight.*

One of the main concerns for the DN30 package, which carried UF_6 material with a maximum 5 percent enrichment of U-235, was the protection of the valve attached to the standard 30B cylinder (designed per the ANSI N14.1 standard), which is made of carbon steel. This valve could be damaged when undergoing HAC drop test conditions; a finite element model of the package using LS-DYNA software was employed to investigate such a possibility. Time history analyses of this package showed inelastic deformations to the 30B tank but not at the valve itself.

With HALEU feed material being more highly enriched than traditional UF_6, it is anticipated that additional design features will be needed for transportation packages in order to meet the regulations. This is exhibited in the DN30-X package, which has additional design features to control criticality and allow for higher enriched UF_6. Specifically, the DN30-X utilizes the 30B-X cylinder, which adds neutron-absorbing materials in a special frame placed inside a traditional 30B cylinder.

Such a frame has to maintain its geometry in order to suppress critically under NCT and HAC drop tests, like baskets used in traditional SNF packages. Like the DN30, valves are connected to the cylinders to support filling of UF_6. Inelastic deformations at these regions of the cylinder are not permitted, as leak rates cannot be verified accurately as a result. Both the DN30 and DN30-X weigh less than 4,536 kg (10,000 lb) and utilize the protective structural package overpack, which has a clam shell closure system, making it easier to handle, secure (tie down), and transport via truck with respect to traditional SNF packages. UF_6 is placed in liquid form and, due to its low triple point, is often treated as a fragile solid. The form of the UF_6, whether liquid or solid, does not pose a particular challenge to the containment boundary of the package from a structural point of view.

Another source of feed material that is not from a UF_6 source is uranium dioxide (UO_2) powder that has been down-blended from HEU. One package that is currently approved to carry this content is the OPTIMUS-L, as described in Eidelpes et al. (2020). However, the OPTIMUS-L is only licensed to carry small amounts of UO_2 (no more than a few kg) and does not have additional criticality controls like the DN30-X to handle larger amounts. Eidelpes et al. (2020) propose a new design to carry larger amounts of UO_2 based on the existing OPTIMUS-L design called the HALEU transportation concept. The concept uses smaller, inexpensive custom-made canisters that fit in a basket that incorporates borated aluminum for criticality control and relies on foam for impact resilience (Figure 1).

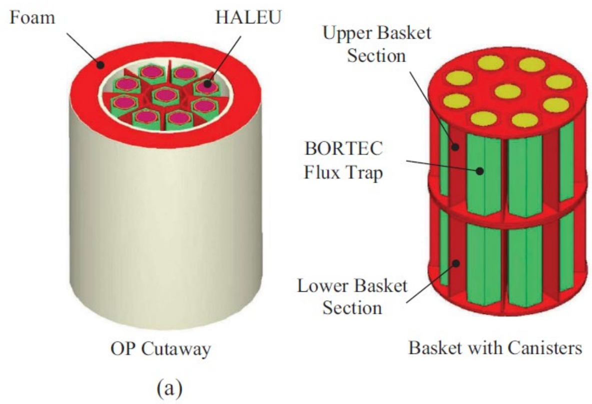

Figure 1. HALEU Transportation Concept (Eidelpes et al. 2020)

As discussed for the DN30-X, the HALEU transportation concept from a structural point of view is more complex than the DN30 since the additional basket structure would have to be analyzed as well.

A package that is certified to transport UF_4 in small quantities is the Versa-Pac (Orano 2022b). UF_4 is a solid at room temperature and will not pose any structural challenges beyond what has already been seen. No specific guidance is provided by NUREG-2216 for UF_4, but a basket used to control criticality is anticipated at large quantities.

Based on previous UF_6 packages designs, NUREG-2216 could be updated to elaborate on the need to protect ports and valves that are common to tanks, like the 30B filled with UF_6, and should elaborate on the physical form of the UF_6 that could damage the containment boundary, such as brittle UF_6 during drop test conditions. No elaboration on highly enriched oxide powders is needed from a structural point of view.

4.3 Thermal

Similar to the structural discipline, the thermal requirements of 10 CFR Part 71 and 49 CFR Parts 100-185 must be met before a package can be certified for the transportation of nuclear and possibly radioactive materials. Packaging definitions used to transport HALEU feed material and HALEU must be capable of withstanding intense thermal environments while preventing reconfiguration or release of the contents and nuclear subcriticality. Achieving this capability requires the use of construction materials that enable the package to withstand serious thermal insult. Accordingly, all Type B, Category 1 safety significant components of a packaging containing HALEU feed material and HALEU must be demonstrated to survive a broad temperature range and significant temperature differential exposure as identified in associated regulations presented in Section 3.0.

Design strategies that account for contents that generate a relatively large amount of heat are not expected to be necessary, as even hundreds or thousands of kilograms of HALEU to fuel a single reactor core stemming from recovery and polishing processes will generate a limited amount of decay heat. As such, the issue of thermal protection remains simpler because insulating material must only work effectively to reduce the heat added to the package during an upset or accident condition instead of having to also consider allowing internally generated heat to escape under regular operating conditions. Thermal management design goals typically conflict with one another where heat-generating content, thermal design requirements, and external heat loads must be carefully balanced during the packaging design evaluation process. Nevertheless, an effective thermal insulation scheme must be evaluated and determined not to cause the interior portions of the package to overheat under HAC or even possibly NCT conditions as overheating could then lead to failures of safety significant items.

It is important that the packaging definition be designed such that all containment items withstand the highest expected temperature under HAC after the entire structural impact loading sequence (NRC 2005b). After the HAC structural and thermal loading sequences of the packaging definition, a maximum leak rate of an A_2 per week is permissible under HAC as specified in 10 CFR 71.51(a)(2).

Both the DN30 and DN30-X used standard 30B cylinders or a slightly modified one such as the 30B-X for the transport of UF_6. A thermally susceptible design feature of these cylinders is the solder that is used to attach the valves to the cylinders, which are filled with UF_6. The solder tends to be quite sensitive to relatively low temperatures, (≈180°C), which if exceeded could cause inadvertent release. Another potential thermal challenge is due to the heat capacity of the UF_6 and the high pressure that can result from a phase change at high temperatures; a partially filled cylinder may be more susceptible to hydraulic failure than a full cylinder.

4.4 Containment

The containment review verifies that the cylinder meets the containment criteria in ANSI N14.5 for Type B packages. Several packages have already been successfully licensed to carry UF_6, UF_4, and UO_2. Relatively larger quantities of UF_6 have been licensed in comparison to UF_4 and UO_2, which have depended on a 30B tank meeting ANSI N14.1 requirements. The containment system for UF_6 HALEU feed material has been typically defined by the 30B cylinders that comply with ANSI N14.1-2019 requirements, which permitted enrichment of only 5 percent. Future designs may incorporate larger tanks as current transportation packages are relatively small, but the containment boundary will still have to resist puncture, water ingress, and thermal loads and meet leakage criteria per NCT and HAC conditions. Thermal loads can cause increased pressure within the tanks and may challenge any solder used on the vent and ports located on a tank. The admissible temperature for materials is often quite low (less than 140°C), and pressure buildup due to melted UF_6 should be noted and should be accounted for when performing immersion test from 10 CFR 71.73(c)(5).

Powder form UO_2 is not expected to cause additional issues with containment as previous packages with powder form UO_2 have been approved.

Shipping HALEU feed material and HALEU contents requires that the package design/definition meet appropriate 49 CFR Part 173, Subpart I, and 10 CFR Part 71 requirements, as well as the recommendations of NRC RG 7.11 (NRC 1991a). Generally, 49 CFR 173.417 permits transportation of fissile material in Type AF, B(U)F, and Type B(M)F packages, and if the fissile material meets the applicable standards in 10 CFR Part 71, a SARP will be required to be prepared and submitted to a regulatory body for all three. By definition, a Type B package is required if the radionuclide inventory exceeds 1 A_2 quantity (49 CFR 173.431). Evaluations performed for transport of fresh (unirradiated) HALEU composed of recovered and polished material determined that approximately only 3.5 kg or more of HALEU is required to exceed one A_2 (Eidelpes et al. 2020). Any one of the proposed commercial advanced reactors will likely require hundreds to thousands of kilograms of HALEU to fuel a single reactor core. As such, the quantities to be transported can be readily characterized as requiring a Type B(U)F or Type B(M)F packaging definition.

RG 7.11 is the first regulatory guidance to introduce the graded approach based on content categories. The graded approach divides the packages into three categories based on the form and activity level of the contents. The package categories (I, II, and III) indicate the ASME BPVC design requirements imposed on the package to ensure adequate design and development of the component safety groups. These include containment, subcriticality/criticality control, and other items important to safety. In this approach, Category I contents have the highest level of activity and require the highest standards and margins of safety.

According to RG 7.11, only 30 A_2 or greater is required to elevate the associated packaging requirements from Category III to Category II. This activity limit is reached by approximately 100 kg of unirradiated HALEU (Eidelpes et al. 2020). This leads to the conclusion that the transportation of HALEU content of the magnitude required for any one of the proposed commercial advanced reactors requires at least Category II containment. Otherwise, content capacity would be well below that needed to support commercial-scale fuel production. However, very little difference exists between Category I and Category II packaging design criteria per the applicable sections of the ASME BPVC per NUREG/CR-3854. To provide flexibility in case the cleanliness of the HALEU is less ideal than anticipated, it is advised that a Type B, Category I packaging design criteria be considered. In this case cleanliness is important to the dose that may be put off by contents and the potential for criticality. Without adopting the Type B Category I packaging definition there is the potential for shipping issues where contents are not within the operating envelope of the package. The sections that follow explain the design and evaluation implications associated with content category in greater detail.

In the case where a specific packaging is used to transport HALEU feed material and HALEU, the packaging system used to ship the content must be designed so that the contents, convenience canisters, compartmentalized shoring, and storage are not subjected to excess shock and vibration that could damage internal components and reconfigure/redistribute or even possibly disperse contents. However, the need to not subject cargo to excess shock and vibration is not unique to HALEU material packaging. Potential sources of shock and vibration data include NNSA SG-100 (NNSA 2005), NUREG/CR-2146 (Fields 1981), and NUREG/CR-0128 (Magnuson 1978). MIL-STD-810H Method 514.8 (DOE 2019) also provides guidance for defining vibration environments to which the material may be exposed throughout a life cycle and for the conduct of laboratory vibration tests.

4.5 Shielding

Feed material is non-irradiated and typically does not need to be shipped in a Type B package and does not require a rigorous shielding evaluation. However, if the feed material is from reprocessed or downblended fuel, product impurities and actinides such as U-232 can cause the package to exceed the Type A package limits and a Type B package may be needed (Eidelpes et al. 2020). Although a shielding evaluation will need to be performed, regulatory dose rate limits in 10 CFR 71.47 and 10 CFR 71.51(a)(2) are typically not challenged.

U-232 has a 68.9-year half-life and decays through a series of much shorter-lived daughter products to the stable nuclide lead-208 (Pb-208). This decay chain goes through thallium-208 (Tl-208), which gives off a particularly strong gamma at 2.6 MeV. Because of this strong gamma radiation at the end of the decay chain, the hazard from U-232 daughter products is dependent on the amount of time that has passed since reprocessing (IAEA 2007).

Chapter 5 of NUREG-2216 discusses reviewing shielding analyses for transportation packages. It states that daughter radionuclides should be addressed and that the source term should be at an appropriate decay time that maximizes radiation levels from parent and daughter radionuclides. Maximizing source from U-232/Tl-208 may not be practical. Figure 2 shows the activity in gammas/sec as a function of gamma energy for several decay times for 1 curie of U-232 from the ORIGEN code. The gamma spectrum is dominated by that of Tl-208. The decay time in this figure that gives the maximum source term for U-232 impurities is 10 years. NUREG-2216 may need to be updated to discuss the presence of impurities, particularly U-232, in a HALEU UF_6 package. One practical application of this would be CoC limitations on decay times after reprocessing. A significant presence of daughter products, such as Tl-208, that are not considered within the shielding evaluation could also be disallowed for shipment.

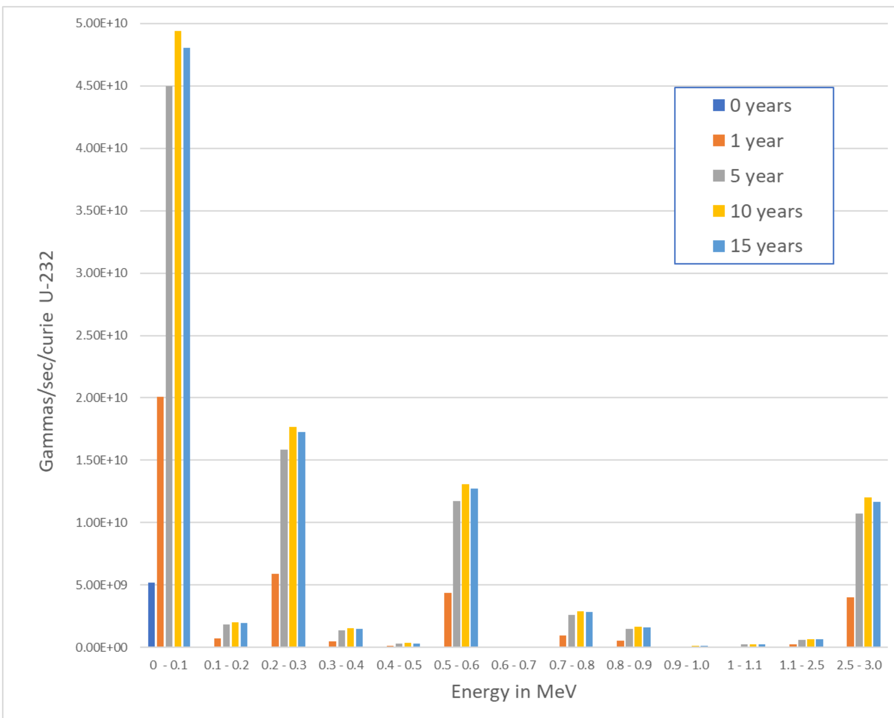

Figure 2. Gamma Activity from the Decay of U-232 from ORIGEN

4.6 Criticality

All performance standards and regulatory requirements for U.S. certification of fissile material packages are prepared by the NRC and provided in 10 CFR Part 71. The NRC and the DOT work together to ensure that the DOT regulations for transporting hazardous material (49 CFR Part 173 and 10 CFR Part 71) are consistent. The performance standards and requirements for certifying packages containing fissile material are only included in 10 CFR Part 71.

For packages that transport fissile material, adequate protection is provided by using a design and safety-assessment philosophy that effectively eliminates the possibility of a criticality event occurring under any and all credible non-transport as well as transport scenarios. This would be no different for applications pertaining to HALEU feed material and HALEU transport. A detailed consideration of the many parameters that interact to influence the neutron behavior is needed to provide an adequate safety basis for the package design definition. Principal parameters that affect criticality safety are 1) type, mass, and form of the fissile material; 2) moderator-to-fissile material ratio (degree of moderation); 3) amount and distribution of absorber materials; 4) package geometry—internal and external; and 5) reflector effectiveness.

NUREG-2216 was reviewed to evaluate potential gaps for enrichments between 5 and 19.75 percent. Previous work (Center for Nuclear Waste Regulatory Analyses 2020) has noted a potential lack of criticality benchmark experiments within the 5 to 19.75 percent range. Saylor et al. (2021) and Zipperer et al. (2020) discuss benchmarks as well as the use of sensitivity and uncertainty analysis methods to identify appropriate benchmarks. The NUREG-2216 process remains the same; however, it is recommended that this reduction in the number of available benchmarks be mentioned within NUREG-2216, *Benchmark Evaluations*, Section 6.4.6. Having fewer benchmarks does not change the process of evaluation but may require additional scrutiny or justification of bias and bias uncertainties. Additional margin in the analyses or the bias uncertainty may be taken to account for the lack of available benchmark data. The reviewer will have to apply expert judgment to collectively evaluate the additional margin if taken, the number of applicable benchmarks, and other conservatisms within the model.

The authors recommend adding enrichment to the list of experimental data in Section 6.4.6.1 of NUREG-2216 at the end of the sentence “verify that the application addresses the overall quality of the benchmark experiments and the uncertainties in experimental data” for reinforcement of enrichment concerns.

Another impact of enrichment greater than 5 weight percent is the exemption from water intrusion. 10 CFR 71.55(g)(4) does not allow exemption from water intrusion or leakage into or out of the containment system since enrichment is over 5 weight percent; therefore, all criteria in 10 CFR 71.55(b) needs to be demonstrated.

Note also that in the enrichment range from 6 to 34 weight percent, heterogeneous systems or models will yield the smallest critical volume, whereas homogeneous systems will yield the smallest critical mass. This differs for enrichments less than 6 weight percent as both the critical volume and critical mass are smallest for heterogeneous systems (Zipperer et. al 2020).

4.7 Materials

The materials evaluation will need to consider mechanical and thermal properties of the package materials. Special attention should be paid to chemical interactions of feed material under the influence of air, water, or both. The DN-30X and the proposed HALEU UO_2 powder concept using the OPTIMUS-L (Eidelpes et al. 2020) rely on neutron absorbing materials for criticality control to be able to transport higher enriched feed materials. NUREG-2216 states that up to 90 percent of the boron in the absorber can be credited within the criticality evaluation if neutron attenuation or other appropriate tests have demonstrated the absorber material is homogeneous. Additional information is available to NRC reviewers in Attachment 7A to NUREG-2216.

4.8 Operating Procedures

An NRC-licensed transportation package is expected to be operated in the manner consistent with its design and evaluation for approval. Operating procedures focus on package loading (including leak testing), unloading (including inspections and tests), empty package for transport considerations (such as decontamination), and other procedures. Regulations specific to these considerations include 10 CFR 71.31(c), 71.35(c), 71.43(g), 71.47(b)(c)(d), 71.87, and 71.89. Additionally, 49 CFR 173.428 may apply. The review of operating procedures ensures that the valve is properly closed and leak-tested, as appropriate, and that the valve protection device, if applicable, is installed. This review also confirms that the radiation levels are verified to meet the regulatory limits prior to transport.

Operations are expected to be very similar to those already existing for packages that are already licensed for UF_6, such as the DN30. However, these packages weigh less than 10,000 lb and can be lifted with a forklift. If larger amounts of HALEU feed material are to be shipped, the package will undoubtedly get heavier, necessitating different equipment that can lift and maintain the package. A heavier package may need trunnions, in which case lifting operations will need to be considered and may need to be rigged differently when tied down to the conveyance. Additionally, the number of packages that can be grouped together in a shipment may also change with increasing the quantity of feed material (criticality safety index [CSI] will need to be verified). However, these additional needs are not expected to be any different from those used for traditional SNF transportation packages, and therefore, NUREG-2216 does not need any major revisions to Chapter 8.

4.9 Acceptance Tests and Maintenance

A certified transportation package has a maintenance program documented in the CoC to assure packaging performance requirements are being met while in service. This includes visual inspections, weld examinations, thermal tests, shielding tests, leak tests according to ANSI N14.5, neutron-absorber and moderators tests, and lifting tests, among others. Before first use, each package is subject to tests that correspond to its design, and periodic maintenance requirements are detailed for replacement of components and repair. The pertinent regulations are 10 CFR 71.31(c), 71.37(b), 71.85(a)(b)(c), 71.87(b)(g), and 71.93(b). The review of the acceptance tests and the maintenance program evaluates the inspection procedures for the overpack, including the physical condition of the inner and outer shells, corrosion, performance of the foam while the overpack is in service, and wear of cushioning pads between the cylinder and overpack. The review also verifies that the cylinder is tested and maintained in accordance with the requirements in 49 CFR 173.420, “Uranium hexafluoride (fissile, fissile excepted, and non-fissile),” and ANSI N14.1. For foam-filled overpacks, the acceptance tests for the foam should include reasonable ranges for material density, compressive strength, thermal conductivity, and other factors. Structural analyses may be used to justify the ranges. Reference to ASTM International standards should be reviewed to ensure that the standard does not overly restrict the testing of foam characteristics.

Based on previous packages that have been approved to transport feed material, new additional acceptance test and maintenance requirements are not expected to be necessary. Experience from fresh fuel and SNF transportation packages are expected to be leaned on when larger shipments of feed material are shipped, and therefore, NUREG-2216 will most likely not need any major revisions in Chapter 9.

4.10 Quality Assurance

QA requirements for HALEU packages will not vary from other fuel feedstock packages. Although the contents may change, there is no substantial change in requirements based off this difference.

4.11 Appendix A

Appendix A to NUREG-2216 has descriptions, safety features, and areas of review for different types of radioactive material transportation packages.

Section A.6 of NUREG-2216 is for LEU hexafluoride packages. This section could be updated to address both low (≤ 5 percent) and higher (≥ 5 percent) enriched uranium. Recommended updates from Sections 4.2 through 4.10 of this report could be added to Section A.6 to address the differences in evaluations in a ≥ 5 percent enriched UF_6 package.

5.0 SSR-6, *Regulations for the Safe Transport of Radioactive Material*, IAEA Safety Standards

The NRC performs revalidation reviews at the request of DOT in accordance with its MOU with the DOT (U.S. Department of Transportation and U.S. Nuclear Regulatory Commission 1979) to ensure that foreign packages conform to IAEA safety standards. This is important for international shipments that may arrive at U.S. ports of entry and then traverse to a U.S. facility. Current 10 CFR Part 71 regulations are in general accord with the 2009 edition of the IAEA's *Regulations for the Safe Transport of Radioactive Material* (TS-R-1) (IAEA 2009). The IAEA has since updated its standards for the transport of radioactive material in *Regulations for the Safe Transport of Radioactive Material*, SSR-6 (2012 and 2018 Editions) (IAEA 2012, 2018). The NRC had published some proposed revisions to 10 CFR Part 71 regulations to be consistent with that of the updated IAEA safety standards (NRC 2022b).

One proposed revision to UF_6 package requirements relates to the cylinder plug. 10 CFR 71.55(g) provides the conditions under which UF_6 packages can claim moderator exclusion. The condition in 10 CFR 71.55(g)(1) states that following the tests specified in 10 CFR 71.73, "Hypothetical accident conditions," there cannot be physical contact between the valve body and any other component of the packaging, other than at its original point of attachment, and the valve must remain leaktight. The 2018 version of the SSR-6 (IAEA 2018) added the same requirement for the cylinder plug. The NRC proposed to update 10 CFR 71.55(g)(1) to have the same requirement. Considering that moderator exclusion is only allowed for 5 percent enrichment or less, this change would not apply to any UF_6 package with enrichment greater than 5 percent.

In addition, a crosswalk between IAEA and 10 CFR Part 71 regulations would help make these reviews more efficient. This lack of guidance is identified as a deficiency in the NRC guidance documents, but it is not specific to HALEU feed materials and UF_6 packages; it extends to all foreign certificate packages for radioactive and fissile material. It is recommended that the NRC create guidance for reviewers that relates NUREG-2216 to all relevant revisions of the IAEA regulations, as package revalidation reviews are performed to the specific version of the IAEA regulations under which the foreign certificate was issued.

McGhee (2014) compares IAEA regulations and 10 CFR Part 71. Since this document is from 2014, the comparison is between the 2012 SSR-6 to the 10 CFR Part 71 regulations at that time. The information from this document should be updated and incorporated into NRC review guidance, and should include other updates to IAEA regulations, such as the 2018 SSR-6 Rev. 1 edition.

Differences between 10 CFR Part 71 and IAEA regulations for UF_6 packages identified in McGhee (2014) are restated below. These are based on the 2012 SSR-6; the preface of 2018 SSR-6 Rev. 1 did not state that there were any changes to these paragraphs, so they are also relevant to the 2018 SSR-6 Rev. 1:

- UF_6 Proper Shipping Name (SSR-6 paragraphs 419 and 425): Table 1 and paragraphs 419 and 425 in SSR-6 allow for UF_6 to be shipped in an excepted package under the new proper shipping name, "UN3507, Uranium Hexafluoride, Radioactive Material, Excepted Package," if the UF_6 is fissile excepted or non-fissile, is less than 0.1 kg, and meets the requirements of paragraphs 420 for packaging and 424 for limited quantities of material. Neither NRC nor DOT include this proper shipping name. Under current DOT regulations, a limited quantity of UF_6 would be classed as a corrosive material.

- UF_6 Packaging (SSR-6 paragraphs 631, 632, and 718): Paragraphs 631 and 632 give requirements for UF_6 packaging, including requirements to comply with the International Organization for Standardization 7195 and a thermal test; however, DOT requirements reference ANSI N14.1. Also, DOT requires a pressure test to 1.4 MPa. Paragraph 718 of SSR-6 requires a pressure test to 1.38 MPa with multilateral approval, or otherwise to 2.76 MPa.
- Excepted Package Requirements (SSR-6 paragraph 422): SSR-6 contains a requirement to the excepted package conditions for less than 0.1 kg of UF_6. This is related to the new UF_6 proper shipping name discussed above, which NRC and DOT do not recognize. However, this packaging requirement is broader than UF_6.

6.0 HALEU Production Status

This section summarizes the current status of the supply of HALEU for use in advanced reactors that may be licensed under NRC jurisdiction, with a focus on the form and chemical composition of the HALEU during transportation activities.

6.1 DOE HALEU Fuel Feed Production Status

A new generation of reactor concepts are in development that use fuel that is in different chemical, isotopic, and geometric forms than found in traditional LWR fuel. This includes liquid fuel-based concepts and concepts using fuel enriched up to 20 percent (e.g., HALEU). Regalbuto (2020) describes different scenarios and drivers as of 2020. As of 2023, there is interest in the supply of HALEU to support the DOE-funded Advanced Reactor Demonstration Projects (ARDPs) as illustrated by statements by Senator John Barrasso, ranking member of the Senate Committee on Energy and Natural Resources, following announcement by TerraPower that attributed a 2-year delay to the TerraPower Natrium Project to unavailability of HALEU (ENR 2022). Investment is being made within the United States to address the supply challenges across the entire spectrum of front-end activities, including mining, conversion to UF_6, and production of HALEU through enrichment and recovery from existing material (WNN 2022). As one example, the DOE Office of Nuclear Energy is working to establish market conditions that encourage private investment in an enduring HALEU enrichment capability through offtake contracts to stock a HALEU bank “as soon as possible” (SAM.gov n.d.; DOE 2022; Nuclear Newswire 2022b). These efforts may lead to a future sustained supply of HALEU originating from enrichment of virgin natural uranium sources from mining, for which a Type A fissile material transportation package can be utilized. This material would be transported in UF_6 form from the enricher to a fuel fabricator under the existing fuel cycle paradigm. However, considering that some proposed advance reactor technologies do not use discrete “fabricated” fuel, HALEU transportation may occur in other forms, such as UF_4, to the reactor site. This shipment would either occur directly from an enricher that deconverts the UF_6 to UF_4 (or other desired form) or from an intermediary that performs the deconversion. There have been several recent developments in the HALEU feed production area:

- The *Inflation Reduction Act* includes $700 million for research, development, and production of domestic HALEU fuel (Huff 2022)
- *Request for Information (RFI) Regarding Planning for Establishment of a Program To Support the Availability of HALEU for Civilian Domestic Research, Development, Demonstration, and Commercial Use* (DOE 2021)
- Centrus signs to complete HALEU demo in 2023 as the DOE prepares draft Request for Proposal (Nuclear Newswire 2022a)
- On December 7, 2022, DOE established the HALEU Consortium (Merrifield et al. 2022)
- DOE Announces Cost-Shared Award for First-Ever Domestic Production of HALEU for Advanced Nuclear Reactors (DOE 2022)

With the target timelines of the ARDP and intense pressure to use U.S.-origin fissile material and supply chains, DOE and industry partners are working to immediately secure HALEU fuel feedstocks before domestic HALEU enrichment capabilities reach large-scale production. This includes obtaining material from DOE derived from downblended HEU declared surplus from other needs. A 2016 accounting of HEU stocks shared by the White House (The White House 2016) summarizes the surplus material available at that time. However, the scenario has evolved, and less material is available for downblend to HALEU and use in civilian nuclear applications (BWXT 2018). Material derived from this source, and not previously identified as spent reactor fuel, is likely to meet requirements for a Type A fissile material transportation package. Historical activities for downblending of HEU to LEU for commercial power reactor use resulted in uranyl nitrate (UN) that was transported to a fuel fabricator. It is not known if downblend to HALEU would be shipped in UN or another form.

An example of material characteristics for HALEU fuel derived from DOE excess material is found with the proposed Microreactor Applications Research Validation and Evaluation (MARVEL) reactor (DOE n.d.). MARVEL is proposed for construction at the Idaho National Laboratory under a DOE authorization basis approval process. The *MARVEL Fuel Fabrication Strategy* (Johnson et al. 2022) describes the source material being in the form of metal scraps from castings obtained from Y-12 that would be transported in this form to the fuel fabricator for conversion and fabrication into fuel slugs. The report summarizes transportation plans to and from the fabricator via a Type B package and includes preliminary chemical analysis of typical HALEU materials from Y-12. The original form (metal scraps) and chemical analysis may be useful to informing other product coming from Y-12 and provided for use in civilian applications. Care must be taken in extrapolating to other materials as Y-12 has material inventory from a wide range of sources.

DOE has proposed to make HALEU fuel available for microreactor demonstration by recycling existing stocks of spent fuel. Baker (2019) (INL/CON-19-54336, *HALEU for Fuel Development and Microreactor Demonstration* [multiple presentations including discussion of recovery of uranium from irradiated EBR-II fuel]) and Patterson (2021) provide background information for this pathway. Technical details for material that may be derived from recycled EBR-II fuel are listed in Baker (2019) (INL/CON-19-54336), including potential chemical and isotopic composition of HALEU fuel emanating from an existing electrometallurgical treatment process applied to the EBR-II fuel. This composition data is informative for what may be expected for HALEU derived from other spent fuel recycling and used in advanced reactor HALEU fuel supply. Care must be taken in extrapolating to the chemical makeup of other materials that may be derived from spent fuel without understanding the type of fuel and reactor it was used in.

6.2 DOE HALEU Fuel Feed Transportation Status

Currently, commercially sized uranium hexafluoride cylinders are limited to enrichments of 5 percent. For example, 30B and 30C cylinders have a capacity of 2,277 kg of uranium hexafluoride but are limited to 5 percent enrichment. Cylinders that are allowed higher enrichments have limited payloads. For example, the 8A cylinder is approved for transport of uranium hexafluoride with an enrichment of 12.5 percent, but its capacity is 115.67 kg of uranium hexafluoride.

Eidelpes et al. (2020) discuss a path forward for the shipment of uranium hexafluoride enriched to greater than 5 percent in commercially sized quantities. For example, a 30B-20 cylinder is discussed that has a goal of being able to transport up to 1,600 kg of uranium hexafluoride enriched to 20 percent.

Eidelpes et al. (2020) also discuss criticality benchmarking, noting that there have been over 5,000 approved International Criticality Safety Benchmark Evaluation Project (ICSBEP) criticality benchmarks, though most uranium experiments are done with less than 5 percent enriched or greater than 20 percent enriched material. This potential lack of experiments in the 5–20 percent enriched range may increase the needed conservatism in package design.

7.0 HALEU Fuel Feed Transportation

7.1 Challenges with HALEU Fuel Feed Transportation

Although HALEU encompasses U-235 enrichment up to 20 percent, typical fuel feed material would be uranium hexafluoride enriched to approximately 8 percent, especially in early applications. This material would be produced at a uranium enrichment facility and shipped to either a deconversion facility or a fuel fabrication facility. The deconversion facility would convert the uranium hexafluoride to uranium metal, uranium oxides, or uranium salts; in the United States, deconversion facilities are typically collocated with fuel fabrication facilities.

One challenge associated with shipping uranium hexafluoride at HALEU enrichments is the availability of commercially sized cylinders, e.g., similar in size to existing 30B cylinders with capacities of 1,590 to 2,270 kg (3,500 to 5,000 lb). Currently, there are no commercially sized transportation packages certified to handle these enrichment levels at those sizes. This is complicated by the fact that uranium hexafluoride at any enrichment is a DOT Class 8 hazardous material (i.e., corrosive), and when uranium hexafluoride contacts moisture in air, it reacts to form hydrogen fluoride and uranyl fluoride, which are extremely corrosive.

Saylor et al. (2021) analyzed the DN-30 uranium hexafluoride transportation package with uranium enrichments of 5.8, 6.7, and 9.5 weight percent and concluded that it would be feasible to ship 30B cylinders with uranium hexafluoride enriched up to 10 weight percent in small arrays. However, DOT regulations (49 CFR 173.420) specify that uranium hexafluoride must be shipped in ANSI N14.1 certified cylinders and the previous revisions of ANSI N14.1 limit enrichments for 30B and 30C cylinders to 5 percent. The 2023 edition of ANSI N14.1 (ANSI 2023) retains this as the standard limit; however, it does allow enrichment above these limits subject to additional criticality assessments and approval by the competent authority, in this case NRC.

A second challenge associated with shipping uranium hexafluoride at HALEU enrichments is the requirement to maintain subcriticality under water intrusion. Transportation of uranium enriched to less than 5 percent in the form of uranium hexafluoride is addressed in 10 CFR 71.55(g), which exempts a fissile material transportation package from the requirement of maintaining subcriticality under water intrusion if the content is uranium hexafluoride enriched to less than 5 percent. Typical HALEU feed material enrichments are in the range of 8 percent, which exceeds the 5 percent limit, and thus, subcriticality under water intrusion needs to be maintained.

A third challenge associated with shipping HALEU feed material is the transport of uranium metal, uranium oxides, or uranium salts from a deconversion facility to a fuel fabrication facility. The primary technical challenge associated with this activity is expected to be the criticality safety assessment associated with the transportation package used for the HALEU feed material. The need for these transportation packages could be eliminated by collocating deconversion and fuel fabrication facilities, as is currently done at LWR fuel fabrication facilities. However, collocating these facilities would be highly dependent on the specific type of fuel that is produced, and it may not always be possible.

A fourth technical challenge is the use of recycled uranium as HALEU feed material. Typically, uranium feed material of any enrichment may be shipped in a Type A fissile material transportation package. However, uranium feed material derived from recycled uranium may require a Type B fissile material transportation package due the presence of U-232, U-234, U-236, and various radioactive impurities, especially transuranic radionuclides. This issue is discussed in Eidelpes et al. (2020). Eidelpes et al. (2020) found that for HALEU feed material derived from recycled uranium from the EBR-II reactor, a Type B fissile material transportation package would be required if more than 3.4 kg of HALEU were to be shipped.

The following subsections discuss key design areas of a transportation package such as structural, thermal, shielding, criticality, containment, and operations, using NUREG-2216 as a guide, and examine the currently licensed transportation packages that carry similar contents to HALEU feed material, such as the DN30, Optimus-L, the Versa-Pac, and those near approval such as the DN30X. Revalidation packages such as the MST-30 package are also examined.

7.2 Future Transportation Packaging Needs

To meet the need for industrial-scale HALEU production, new packages that can transport UF_6, fresh fuel, and other uranium forms at scale are needed because none currently exist that are certified for this type of material at this time. The following section identifies these packaging needs based off of different process flows.

Figure 3 shows the uranium feed material flow as it currently occurs. Uranium hexafluoride is transported from conversion facilities to enrichment facilities to fuel fabrication facilities. At the fuel fabrication facilities, deconversion of the uranium hexafluoride occurs and uranium oxide LWR fuel is fabricated. No new transportation packages are necessary in this scenario. Also note there is no need for a package to ship uranium powder or metal since the deconversion and fuel fabrication facilities are collocated.

Figures 4 and 5 illustrate scenarios where large-scale production of HALEU occurs at enrichment facilities and the HALEU is subsequently transported to deconversion and fuel fabrication facilities. If the deconversion and fuel fabrication facilities are consolidated (see Figure 4), two new transportation packages would be required for large-scale, commercial transport of UF_6 and fresh fuel, which could be in the form of uranium nitride, uranium silicide, uranium metal alloys, or uranium salts. If the deconversion and fuel fabrication facilities are separate (see Figure 5), three new transportation packages would be required for large-scale, commercial transport of UF_6, UO_2 powder, and fresh fuel, which could be in the form of uranium nitride, uranium silicide, uranium metal alloys, or uranium salts. The additional packages are required for the large-scale, commercial transport of uranium enriched to greater than 5 percent.

Figures 6 through 11 illustrate scenarios where LEU enriched to greater than 5 percent (LEU+) or HEU is used to provide a stop gap source of uranium for downblending until large-scale, commercial enrichment capacity for HALEU is developed. This stop gap scenario could potentially provide on the order of a few tens of metric tons of HALEU to demonstrate initial advanced reactor concepts.

Figure 6 illustrates a scenario where stop gap LEU+ or HEU is transported and used to increase the enrichment of uranium from commercial uranium enrichment facilities that is enriched to less than 5 percent. In this scenario, up to three new transportation packages would be required, one package to transport the LEU+ or HEU; another package to ship uranium metal, uranium oxide, or uranium salts from deconversion facilities to fuel fabrication facilities; and a third package to ship fresh fuel in the form of uranium nitride, uranium silicide, uranium metal alloys, or uranium salts.

Figure 7 illustrates a variation on Figure 6, where the deconversion and fuel fabrication facilities are collocated. This decreases the need for new transportation packages from three to two.

Figure 8 illustrates a second variation on Figure 6, where the source facility for the LEU+ or LEU, deconversion facilities, and fuel fabrication facilities are collocated. This decreases the need for new transportation packages from three to one.

Figure 9 illustrates a third variation on Figure 6, where the source facility for the LEU+ or LEU, the blending facilities, deconversion facilities, and fuel fabrication facilities are separately located. This increases the need for new transportation packages from three to four.

Figures 10 and 11 illustrate a fourth and fifth variation on Figure 6, where the source facility for the LEU+ or LEU and the blending facilities are combined, and the deconversion facilities and fuel fabrication facilities are either combined or separate. In these scenarios, either two or three new packages would be required.

As demonstrated by these figures, the need for new packaging may be reduced by collocating facilities as part of the development of a fuel production cycle to support commercial-scale nuclear power production from advanced reactors. Additionally, some packages could be designed or adapted from existing designs to accommodate higher percentages of enrichment, such as to support downblending from HEU, that could then be used for similar fuel form feedstock with lower levels of enrichment (HALEU). Section 7.3 discusses existing packages that could be adapted or utilized.

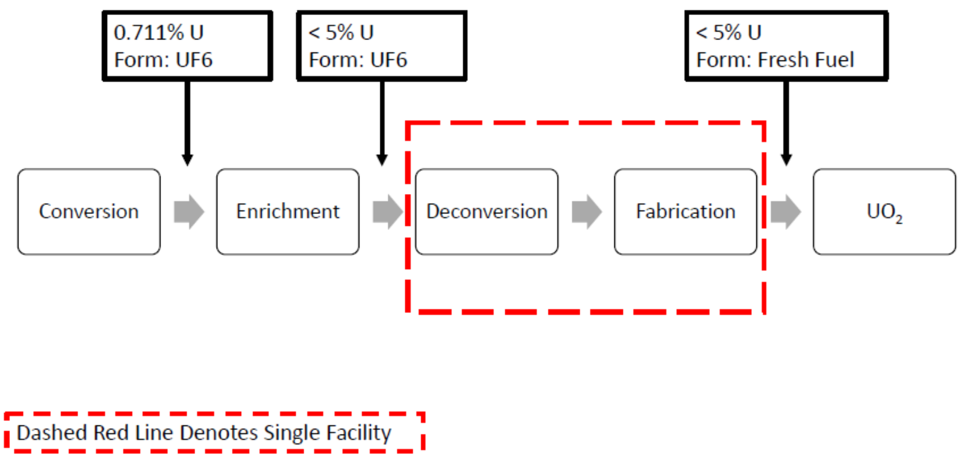

Figure 3. Current Uranium Feed Material Flow

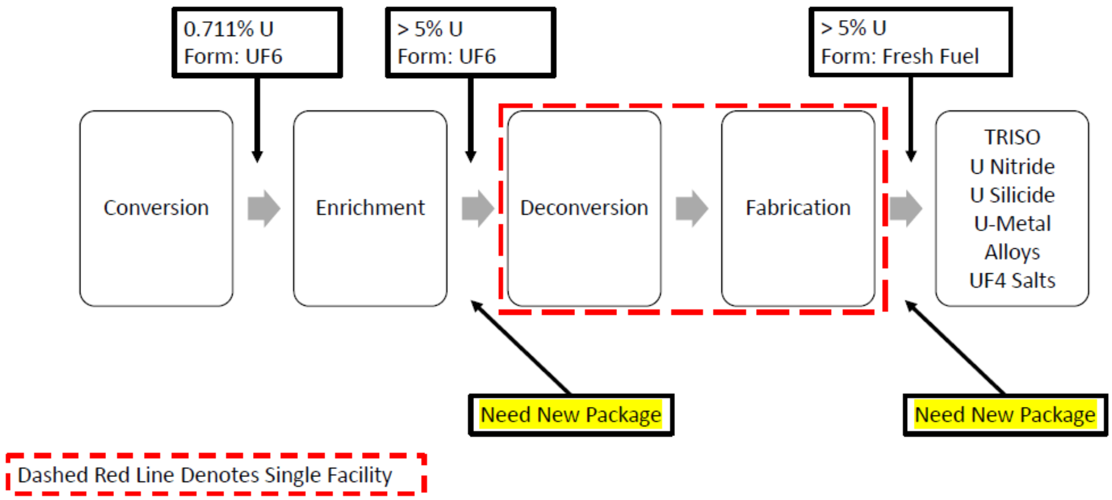

Figure 4. HALEU Production at Enrichment Facilities

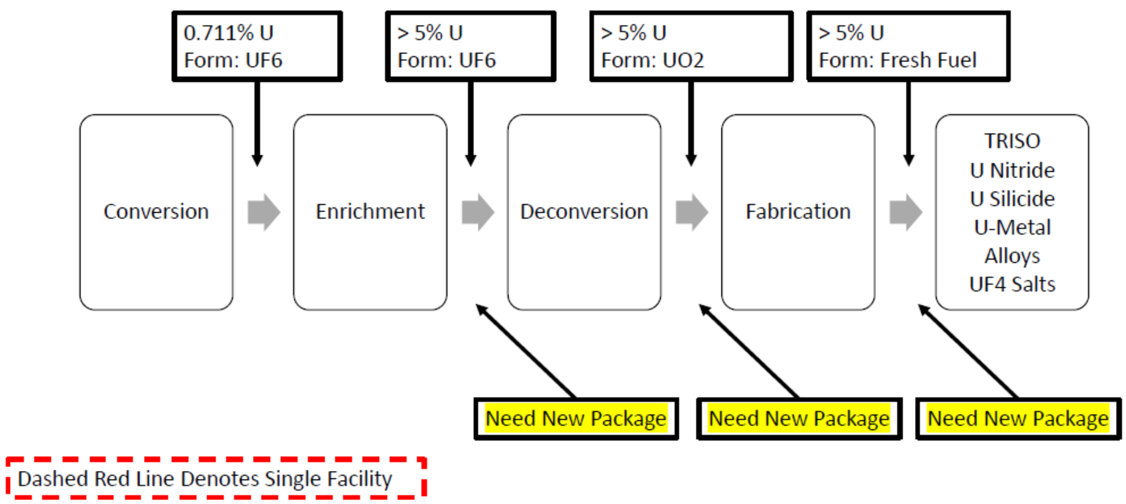

Figure 5. Variation on HALEU Production at Enrichment Facilities

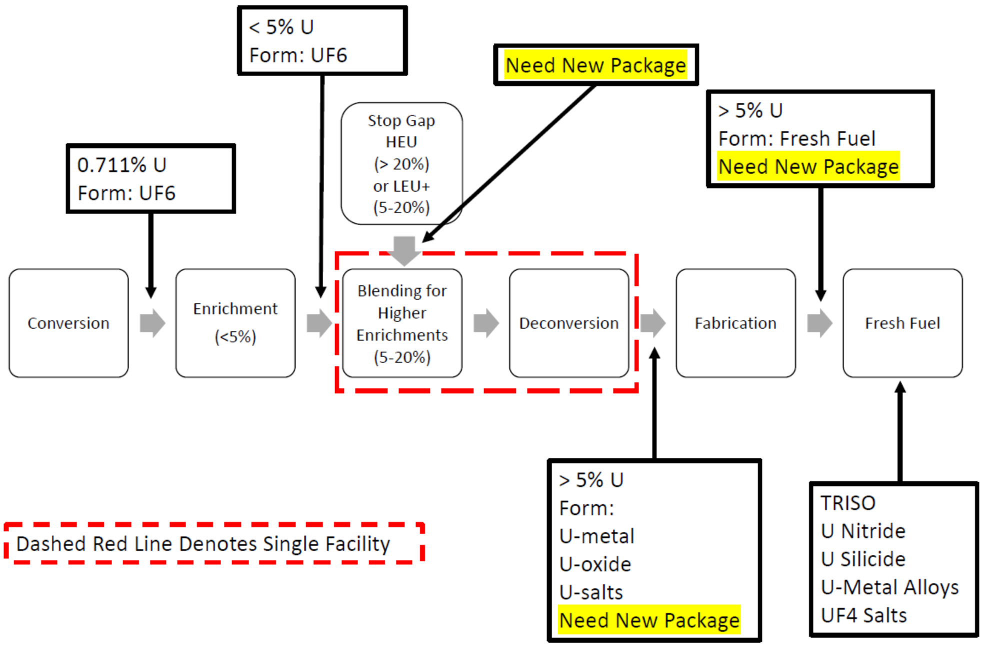

Figure 6. Stop Gap Scenario for Production of HALEU

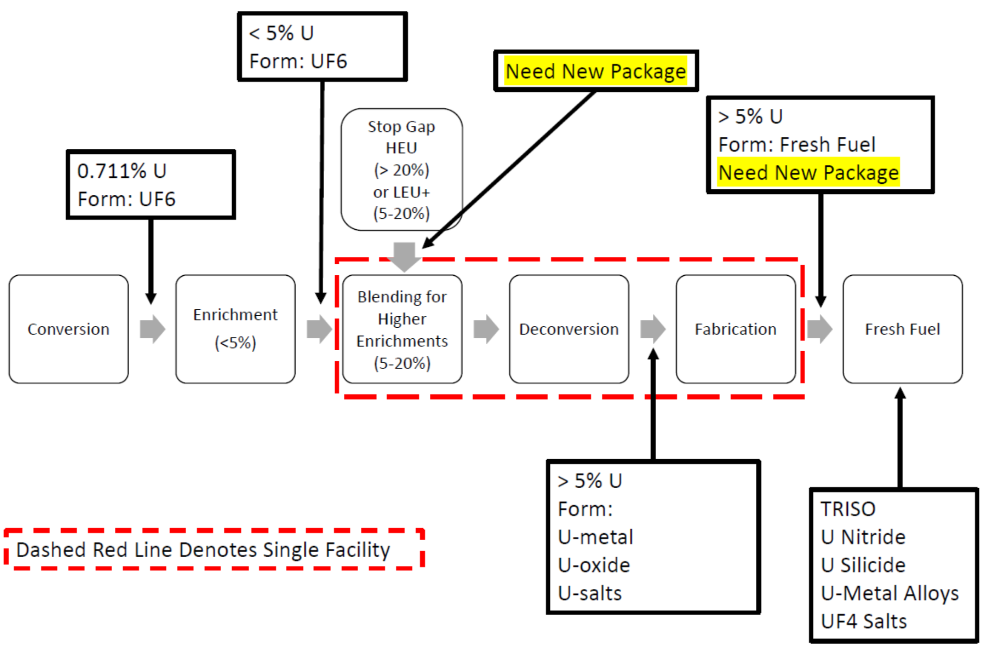

Figure 7. Variation on the Stop Gap Scenario for Production of HALEU

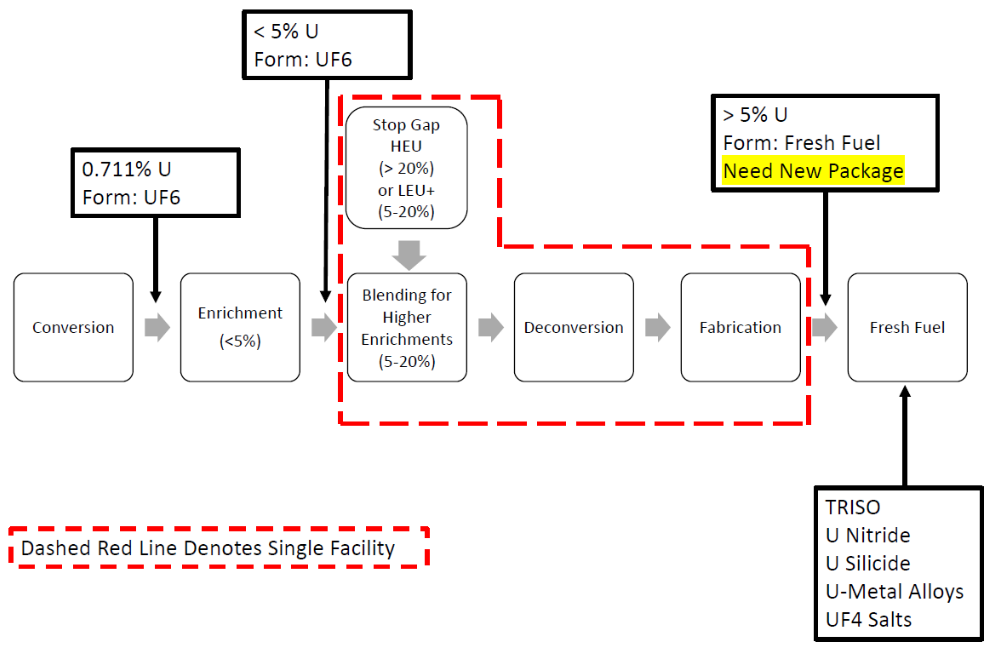

Figure 8. Second Variation on the Stop Gap Scenario for Production of HALEU

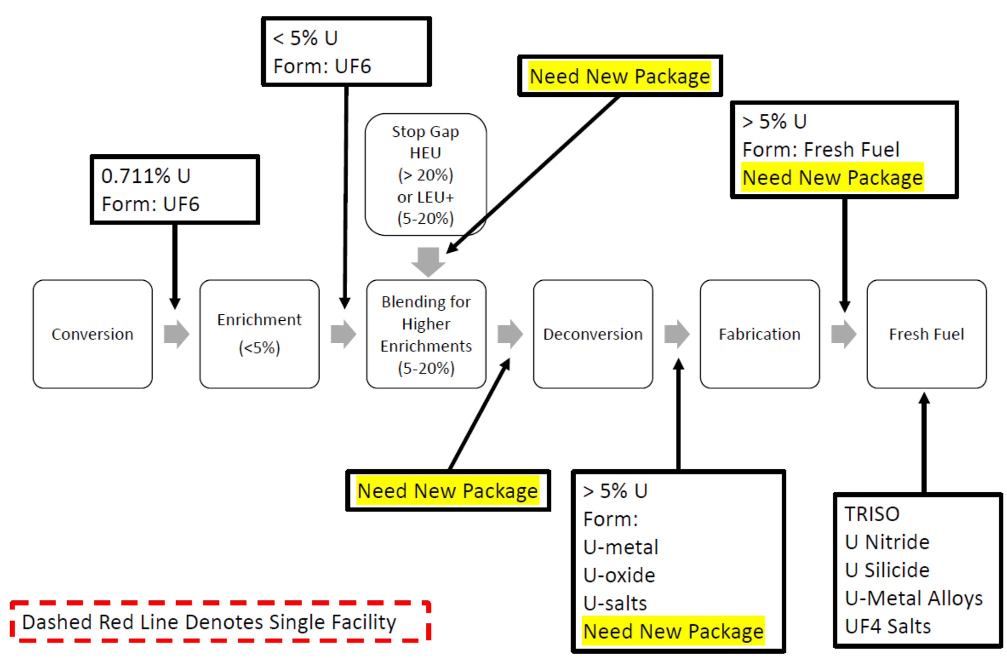

Figure 9. Third Variation on the Stop Gap Scenario for Production of HALEU

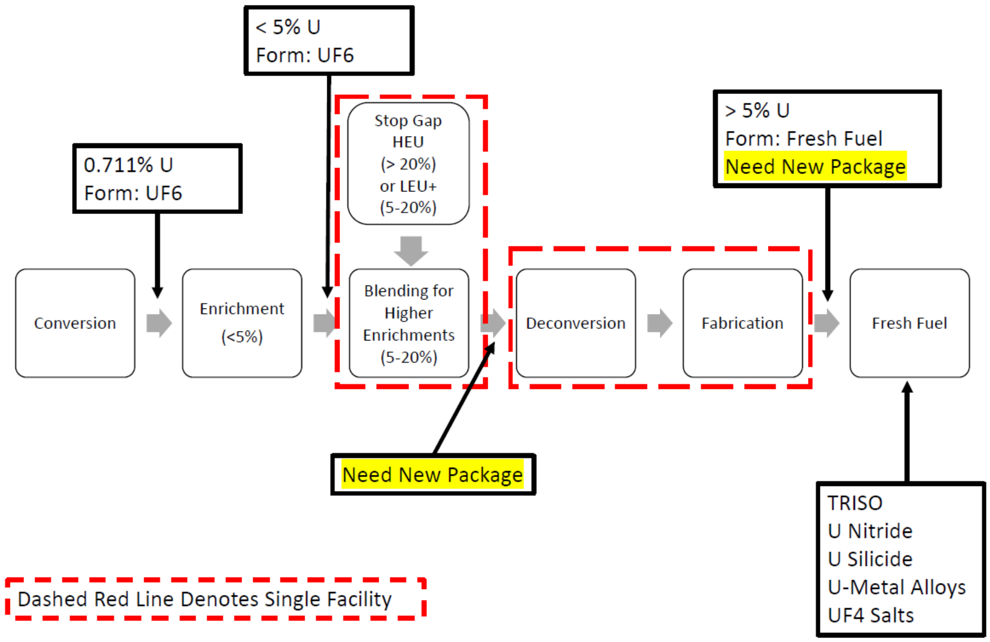

Figure 10. Fourth Variation on Stop Gap Scenario for Production of HALEU

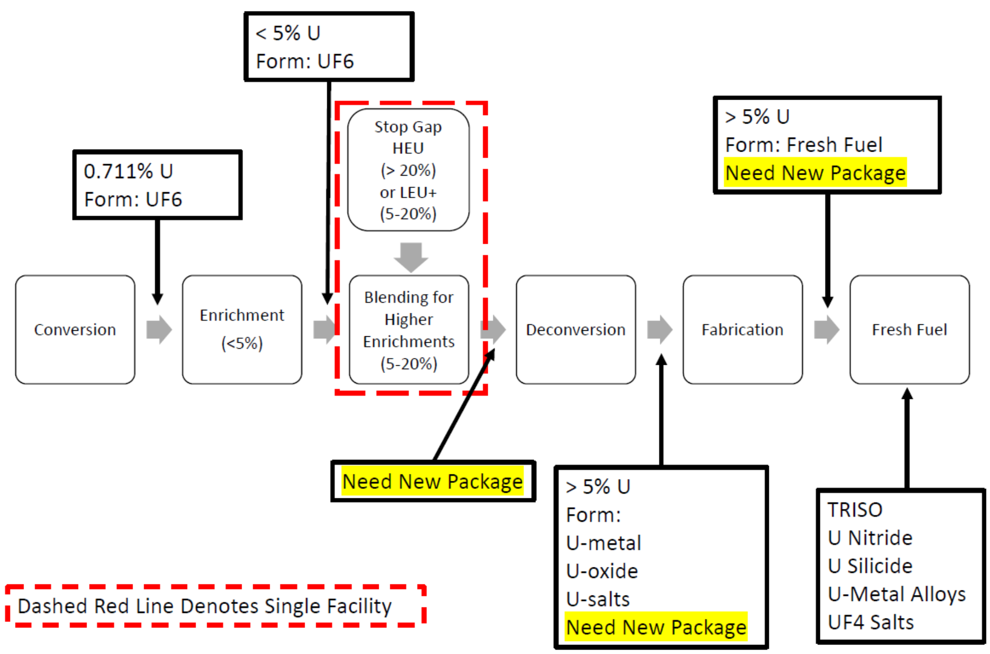

Figure 11. Fifth Variation on Stop Gap Scenario for Production of HALEU

7.3 NRC-Approved Transportation Packages

7.3.1 Packages with Current CoCs

7.3.1.1 UF_6 Packages

Two UF_6 packages that have been issued a CoC by the NRC are the UX-30 (Docket No. 71-9197) (NRC 2018) and the DN-30 (Docket No. 71-9362) (NRC 2021a).

The UX-30 is a Type B package authorized to transport small amounts of impurities from reprocessed fuel that exceed the A_2 limit for a Type A package. The UX-30 is allowed to transport unirradiated uranium with a U-235 enrichment up to 5 weight percent in an ANSI N14.1 Standard 30B or 30C cylinder. It is also allowed to transport reprocessed uranium with a U-235 enrichment not to exceed 5 weight percent. The most recent CoC is Revision 30, dated November 2, 2018.

The DN-30 is a Type AF package and is authorized to transport fresh UF_6 with a U-235 enrichment not to exceed 5 weight percent in the standard 30B cylinder. The most recent CoC is Revision 3, dated August 31, 2021.

ANSI N14.1 states that the maximum fill limit of the 30B and 30C cylinders is 2,277 kg (5,020 lb) UF_6. Table 4 of the standard states that enrichments above 1% require moderation control. The criticality safety analyses for the UX-30 (Columbiana Hi Tech 2018) and DN-30 (DAHER 2019) do not include moderators or water ingress into the containment system. It was determined that the 30B and 30C cylinders contain special design features that would ensure that water could not leak into the containment system so that they qualified for moderator exclusion as allowed by 10 CFR 71.55(c). The packages also met the additional requirements in 10 CFR 71.55(g) for UF_6 packages that employ moderator exclusion, including a limit of uranium enrichment of 5 weight percent U-235.

The CSI controls the number of packages that can be shipped in a single conveyance. Per the requirements in 10 CFR 71.59(c) for a nonexclusive use conveyance, the sum of all CSIs is limited to 50, while for an exclusive use conveyance, the sum of all CSIs is limited to 100. Table 3 summarizes the CSIs and the number of packages that can be transported in a single conveyance for the UX-30 and DN-30.

Table 3. CSI and Number of Packages That Are Allowed in a Single Conveyance for the UX-30 and DN-30

			Number of Packages Allowed in Single Conveyance	
Package	Cylinder	CSI	Nonexclusive	Exclusive
UX-30	30B	5.0	10	20
	30C	0.0	No Limit	No Limit
DN-30	30B	0.0	No Limit	No Limit

The Orano-TLI Versa-Pac is authorized to ship UF_6 in 1S or 2S cylinders at enrichments up to 20 percent, but in small quantities (approximately 28 kg). The 1S and 2S cylinders are designed to be ANSI N14.1 compliant. The most recent CoC for the Versa-Pac is Revision 17, Docket No.

71-9342 (NRC 2010). Table 4 and Table 5 below repeat the CoC tables describing the allowable contents for the 1S and 2S cylinders.

Table 4. Versa-Pac VP-55 1S and 2S Cylinder Limits for UF_6 Enrichment up to 20 Percent U-235

Cylinder Type	Mass UF_6 per VP-55 (g)	Number of Cylinders	U-235 Mass Limit per VP-55 (g)	CSI
1S	3,175	7	429.8	1.0
2S	4,445	2	600.8	1.0

Table 5. Versa-Pac VP-55 1S and 2S Cylinder Limits for UF_6 with 5-inch Pipe, U-235 Enrichment up to 100 Percent

Cylinder Type	Mass UF_6 per VP-55 (g)	Number of Cylinders	U-235 Mass Limit per VP-55	CSI
1S	454	1	306	1.0
2S	2,223	1	1497	1.0

Transport of LEU as UF_6 in 30B and 30C cylinders in UX-30 and DN-30 packages arguably sets the precedent for commercial operations. These packages allow up to 2,277 kg of UF_6 to be transported but are not currently authorized to ship uranium with an enrichment higher than 5 percent. In contrast, the Versa-Pac VP-55 packages are authorized to transport uranium with an enrichment up to 100 percent, but the mass can be no more than approximately 0.2 percent of that allowed for LEU. The authors consider that a package supporting commercial operations should have a capacity within an order of magnitude of the commercial precedent set by LEU transport.

The DN30-X (NRC Docket No. 71-9388) was approved by the NRC in March 2023 for transportation of UF_6 with enrichments up to 20 percent. The safety analysis report (Orano 2022a) states that there are two variants of the package: the DN30-10 and the DN30-20. Table 6 summarizes the proposed allowable contents.

Table 6. Summary of Proposed DN-30X Contents

Package Variant	Cylinder	Enrichment Limit	Mass of UF_6 in kg	CSI
DN30-10	30B-10	10	1460	0
DN30-20	30B-20	20	1271	0

The 30B-10 and 30B-20 cylinders are variants of the standard 30B cylinder but include an interior criticality control system, which consists of control rods containing a neutron poison in the form of B4C and lattice holders that keep each control rod in place. Because the enrichment exceeds 5 percent, the DN-30X cannot apply moderator exclusion, and therefore, the criticality analyses assume that there is water in-leakage to the most reactive extent. Authorization of these packages provides for commercial operations given the UF_6 mass limit is within an order of magnitude of the commercial precedence (2,277 kg LEU UF_6).

7.3.1.2 Packages that Could Potentially Ship Higher Enriched UO_2 Powders and Compacts

Versa-Pac

As well as uranium hexafluoride, the Versa-Pac is also authorized to ship HEU powder and uranium compounds such as UF_4. This content is allowed in both the VP-55 and VP-110. The limits are based on the enrichment limit, how much hydrogenous packing material is present, whether the material will be inside the 5-in. pipe, and whether the package will be transported by air or by ground/vessel transport. In addition, there are different CSIs for different allowable loadings. As discussed in Eidelpes et al. (2020), highly enriched UO_2 powder from EBR-II fuel could be used to make HALEU. Because the Versa-Pac is a Type AF package and does not allow for actinides and impurities in the amount exceeding 1 A_2 (see Section 3.0), this package may not be able to transport large quantities of this material. Table 7 shows the U-235 mass limits for various conditions. These limits apply to packages with no limit on hydrogenous packing materials. Table 8 has higher U-235 mass limits for when hydrogenous packing materials are limited to 454 g (1 lb).

To conclude from the tables, neither the VP-55 nor VP-100 has the capacity to ship commercial-scale quantities of HALEU, assuming the LEU UF_6 mass limit of 2,277 kg sets the commercial precedent. The transportable mass of these packages is limited to approximately 7 kg of total uranium at 10 percent enrichment. Even accounting for UO_2 being 76 percent of the mass of UF_6 for the same uranium mass, these packages are several orders of magnitude too small to support commercial operations.

Table 7. Versa-Pac VP-55 and VP-100 Loading Limits for Uranium Materials (excluding UF_6)

Weight Percent U-235	U-235 Mass Limit (g) for VP-55 and VP-110		CSI	U-235 Mass Limit for VP-55 with 5-inch pipe (g)		CSI
	Ground/Vessel	Air		Ground/Vessel	Air	
≤ 100%	360	360	1.0	695	395	1.0
≤ 20%	445	445	1.0	1,215	495	1.0
≤ 10%	505	505	1.0	Limited by volume of pipe, 122kg U-metal, 60 kg UO2, 45 kg U3O8	590	0.7
≤ 5%	610	610	1.0	Limited by volume of pipe, 122kg U-metal, 60 kg UO2, 45 kg U3O8	790	0.7
≤ 1.25%	1,650	--		Limited by volume of pipe, 122kg U-metal, 60 kg UO2, 45 kg U3O8	790	0.7

Table 8. Versa-Pac VP-55 Loading Limits for Uranium Materials with Limited Hydrogenous Packing Material (Excluding UF_6)

Weight Percent U-235	U-235 Mass Limit (g) for VP-55		Material in 5-inch Pipe Container for VP-55[(a)]	
	CSI=0.7	CSI=1.0	No. pipes	CSI
≤ 100%	515	--	1	395
≤ 20%	605	635	2 in high-capacity basket (U-metal not allowed)	CSI=0.7 for U3O8, UO3 & UF_4 CSI=1.4 for all other compounds
≤ 10%	685	--	2	CSI=0 for uranium oxides CSI=1.4 for all other compounds & U-metal
≤ 5%	800	--	2	CSI=0 for uranium oxides CSI=1.4 for all other compounds & U-metal

(a) Note that when transporting within the 5-inch pipe container, the mass is limited by the volume of the pipes, which corresponds to mass limits of 122 kg U-metal, 60 kg UO_2, and 45 kg U_3O_8 per pipe.

OPTIMUS-L

A concept has been developed for transporting larger amounts of higher enriched UO_2 powder downblended from HEU (Eidelpes et al. 2020) that uses the Nuclear Assurance Company (NAC) *Optimal Modular Universal Shipping for Low-Activity Contents OPTIMUS-L* (NRC Docket No. 71-9390) (NRC 2022a). Eidelpes et al. (2020) suggest the package capacity could be up to 376 kg of HALEU. This capacity is equivalent to approximately 500 kg of UF_6 and so likely could support commercial operations given the commercial precedence of 2,277 kg UF_6 as LEU described earlier.

For the OPTIMUS-L, the latest CoC is Revision 1, issued on January 24, 2022. It expires on December 31, 2026. The contents for this revision of the CoC do not include the basket concept discussed in Eidelpes et al. (2020) and needed to transport HALEU. This package is currently authorized for transuranic waste and LEU less than 1% enrichment. Although the U-235 limits are far below that evaluated in the concept in Eidelpes et al. (2020), the other package evaluation areas for the currently approved design that are independent of the basket design and contents are likely applicable for the basket design and content in Eidelpes et al. (2020). It is unlikely that a new thermal evaluation would need to be performed, and the mass of the contents in Eidelpes et al. (2020) is bounded by the allowable mass in the current CoC; therefore, the structural analysis of the overpack should be applicable. Although a shielding evaluation may need to be performed for the new contents, it is unlikely that the fresh fuel contents would result in a higher dose rate than the currently approved contents.

CHP-OP-TU

The TN CHP-OP-TU is authorized to ship uranium oxide pellets, powder, and uranium-bearing materials limited to a U-235 enrichment up to 5 percent. The most recent CoC for the CHP-OP-TU is Revision 12, Docket No. 71-9288, issued on July 13, 2020 (NRC 2020a). The CHP-OP-TU is a cube-shaped package. The UO_2 is contained in four steel oxide vessels (OVs) in three available sizes: 8 in., 7.5 in., and 6 in. nominal inside diameter. The mass of all contents is restricted to 729 kg (1,608 lb) per package and 182 kg (402 lb) per each OV. Hall et al. (2020) analyzed this package (and other existing transportation packages) for use with HALEU fuel. Results of this study show that the CHP-OP-TU could be adapted to transport increased enrichment UO_2 powder by reducing the size of the HAC array (i.e., increasing the CSI) and/or reducing the OV diameter. Enrichment up to 18 weight percent U-235 was studied.

ES-3100

The ES-3100 is authorized to ship "[u]ranium as oxide, which may include UO_2, UO_3, and U_3O_8, packaged in stainless-steel, tin-plated carbon steel, or nickel-alloy convenience cans, or polyethylene bottles. The physical form of all contents is dense, loose powder which may contain clumps and pellets." The ES-3100 is also authorized to ship uranium in other forms. The most recent CoC for the ES-3100 is Revision No. 16 Docket No. 71-9315 and was issued on January 5, 2021 (NRC 2021a). For UO_2 powder, the ES-3100 is not limited by enrichment but is limited by mass of U-235, either 9.682 kg U-235 and 921 g carbon with a CSI of 0.0 or a U-235 mass of 12.32 kg, and no carbon with a CSI of 0.4. Although this package is a Type B package and could be used to ship higher enriched UO_2 powder, it is limited in size, as the overall dimensions are 110 cm (43 in.) in height and 49 cm (19 in.) in diameter and its capacity is approximately 100 times smaller than that set by commercial precedence (2,277 kg of LEU UF_6).

Other Packages

The OPTIMUS-L was selected for the design in Eidelpes et al. (2020) because this model provides sufficient shielding while having a lower mass and smaller geometry than other packages to facilitate package operations. Other packages may be considered for including a basket with canisters similar to the design in Eidelpes et al. (2020), such as the EnergySolutions 8-120B (NRC 2022a).

7.3.1.3 Fabricated Fuel Packages

The HALEU nuclear fuel assembly ready for loading into an advanced nuclear could be in several configurations. For example, TRISO fuel can be in the form of either pebbles or prismatic blocks containing channels loaded with fuel compacts. HALEU metal fuel would be more like traditional LWR LEU assemblies consisting of pins containing pellets or slugs.

The Westinghouse Traveller (Docket Number 71-9297) is a shipping package designed to transport non-irradiated uranium fuel assemblies or rods with enrichments up to 5.0 percent. It will carry several types of pressurized water reactor fuel assemblies and either boiling water reactor or pressurized water reactor rods. Notwithstanding the enrichment limit, the configuration of advanced HALEU nuclear fuel types makes the Traveller inapplicable.

7.3.2 Packages Currently Under Review

Notwithstanding the concept outlined above, the NRC is actively reviewing application of the OPTIMUS-L for transporting TRISO compacts containing HALEU. The safety analysis report was submitted by NAC in October 2022. However, the package would only be authorized to carry 68 kg of HALEU in the form of TRISO compacts because it is designed specifically to support near-term microreactor applications. Therefore, the package is of insufficient capacity to support larger-scale small modular reactor applications as measured by the LEU UF_6 metric.

7.3.3 Revalidations

The DOT has issued a certificate for import and export shipments using the Japanese certificate J/159/AF-96 Rev. 3 for the MST-30 package (U.S. DOT 2021). The DOT certificate number for this package is USA/0585/AF-96 Rev. 5 and the NRC docket number is 71-3057. This certificate states that there is up to 2,277 kg UF_6 allowed with a CSI of 0 and an enrichment of 5 percent or less. The NRC SER (NRC 2021b) recommending revalidation of this certificate states that it uses the 30B cylinder. The DOT certificate was issued on February 10, 2021, and expires on March 4, 2025.

8.0 Conclusions and Recommendations

This report discussed the framework and review process for fuel feedstock packaging and transportation. The NRC regulations and guidance were evaluated in detail in Section 4.0. In Section 7.0, the transportation needs are evaluated along with current packaging available or under review. A thorough understanding of the market is important to NRC so that the organization can plan for reviews through resource allocation and prioritization of guidance updates and research. Some key conclusions, and recommendations for the NRC from this research and evaluation:

- NRC should be aware that the 5 percent enrichment threshold poses a significant challenge to UF_6 cylinder designs due to the need to account for moderator intrusion above 5 percent. This requires more rigorous design and review for the structural, containment, and criticality evaluations. This 5 percent threshold will also introduce potential challenges to existing facilities and operations. Most impactful will likely be criticality control through spacing and material loading limits. This will result in new applications for licenses and more in depth reviews required of those licenses.
- The performance-based requirements of 10 CFR Part 71 and related NUREG-2216 guidance are inherently adaptable and do not require any substantial changes because of this adaptability. NUREG-2216 may need minor updates for specific design features of HALEU packages to account for the necessary criticality evaluations. Although an update of NUREG-2216 is not strictly necessary, work now could streamline the review process in the future.
- There are several packages both licensed and under review that can be used to transport HALEU feedstock. One package for transporting HALEU UF_6 was approved this year and is of sufficient capacity to support commercial operations. There are active concepts and under review but none are approved for HALEU powder transportation, and only a concept based on the NAC Optimus-L is of sufficient capacity to support commercial operations. However, transport of powder would only be required if deconversion and fuel fabrication facilities are not collocated (such facilities are collocated for commercial LEU operations). There are no packages appropriate for transporting fabricated HALEU fuel meaning the NRC should expect new license applications for this purpose.
- HALEU encompasses a wide range of enrichments, from 5 to 20 percent, and the review of transportation regulations and guidance did not identify any “cliff edge” technical barriers to increasing enrichment all the way through this range. However, not all reactors will require the upper end of the HALEU range, and in general, package design difficulty will increase as enrichment limits are increased. For this reason, it may not be economical for vendors to license a package at enrichments all the way through the range 5–20 percent. Practically, many shipments will be in the vicinity of 10 percent enrichment. For this reason, the NRC should expect initial packages to not include the fill enrichment range and potentially need reviews to be relicensed or replaced to accommodate higher enrichment.
- More development for criticality benchmarks will be needed to accommodate higher enrichments. Although there is currently no safety issue, the vendors will have difficulty pushing to enrichments approaching 20 percent due to the uncertainty that needs to be incorporated into criticality calculations. For the NRC this could affect certification timelines and review processes. Careful study of the benchmarks will be needed for reviews, and it is possible that vendors will initially certify at lower enrichments and move up the range later.

- The authors recommend that staff review guidance be developed for revalidation reviews to support DOT. International shipments in accordance with SSR-6 may become more common, and NRC reviewers need to understand the differences between SSR-6 and 10 CFR Part 71, specifically for reviewing UF_6 packages.

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