Nuclear Security Impacts of HALEU Fuels in Advanced Reactors

S. Sarnoski[1], S. Bays[2], D. Colameco[3], R. Cumberland[4], R. Fielding[2], K. Geelhood[3], T. Harrison[3], L. Jollay[5], V. Mehta[1], R. Montgomery[4], G. Reyes[2], C. Robinson[6], T. Zipperer[3]

[1] Los Alamos National Laboratory
[2] Idaho National Laboratory
[3] Pacific Northwest National Laboratory
[4] Oak Ridge National Laboratory
[5] Boston Government Services
[6] Y-12 National Security Complex

Abstract

With the anticipated development and deployment of several advanced reactors designed to use High-Assay Low Enriched Uranium (HALEU) fuel, the demand and international usage of HALEU fuel is expected to significantly increase. HALEU is defined as material enriched between 5 and 20wt% ^{235}U and Low Enriched Uranium (LEU) is material enriched to less than 5wt% ^{235}U. It is important to understand any associated security concerns with the increased production and international usage of HALEU. The United States Nuclear Regulatory Commission classifies HALEU fuel facilities as Category II: moderate strategic significance, versus LEU fuel facilities, which are categorized as Category III: low strategic significance. This difference in designation corresponds to more restrictive security requirements for nuclear facilities producing and handling HALEU fuel.

This report investigates the nuclear security impacts of increased use of HALEU and compares them to those of LEU. Specifically, consequences of facility sabotage and radiological sabotage were examined for steps in the nuclear fuel cycle, including enrichment, downblending, fuel fabrication, transport, reactor operations, and irradiated fuel storage and transportation.

1 Introduction

This study investigates the nuclear security impacts of HALEU relative to LEU for key steps in the nuclear fuel cycle, focusing on facility sabotage and radiological sabotage consequences. The fuel cycle steps evaluated include enrichment, fuel fabrication, reactor operation, spent fuel management, and transportation. The study focused on two general advanced reactor designs proposing to use HALEU fuel: a reference sodium cooled fast reactor and a reference helium-cooled TRISO pebble bed reactor. They are compared with a light water reactor (LWR) as an LEU baseline. The assessment uses estimates for the advanced reactor variables; since no final designs have been released, the values applied within this report should be assumed to be correct within an order of magnitude.

2 Nuclear Material Regulations

Differences in nuclear security for HALEU compared to LEU are driven by domestic regulations and international guidance. Current regulations and guidance that mention uranium, mass, enrichment, or form are relevant, and their relevance to HALEU are discussed.

The United States Nuclear Regulatory Commission (NRC) has three categories for nuclear materials based on their assessment of its strategic significance, shown in Table 1. The International Atomic Energy Agency (IAEA) classifies uranium as either highly enriched uranium (HEU) or LEU for safeguards purposes, where HEU references enrichments higher than 20wt%.

Table 1. NRC Categorization of Nuclear Material [1].

Material	Form	Category I	Category II	Category III
Pu		≥ 2 kg	> 500 g	≥ 15 g
^{233}U		≥ 2 kg	> 500 g	≥ 15 g
^{235}U	≥ 20% ^{235}U enrichment	≥ 5 kg	< 5 kg, but >1 kg	>15 g, ≤1 kg
	≥ 10 but <20% ^{235}U enrichment		≥ 10 kg	<10 kg, >1 kg
	<10% but >natural ^{235}U			≥ 10 kg
Or Formula Quantity*		≥ 5 kg	< 5 kg, but >1 kg	≥ 15 g

**Formula Quantity is grams = (grams contained ^{235}U) + 2.5 (grams ^{233}U + grams plutonium)*

The three categories of facilities established by the U.S. NRC are:

- Category I: high strategic significance
- Category II: moderate strategic significance
- Category III: low strategic significance

Currently, there are three Category III fuel fabrication plants in the United States that are processing low-enriched uranium: Global Nuclear Fuel-Americas in Wilmington, NC; Westinghouse Columbia Fuel Fabrication Facility in Columbia, SC; and Framatome, Inc., in Richland, WA. There are two Category I fuel fabrication plants: Nuclear Fuel Services in Erwin, TN and BWXT Nuclear Operations Group in Lynchburg, VA. As of April 2023, there are no operating Category II facilities in the United States.

Based on the current NRC guidance, HALEU fuel fabrication facilities will be categorized as Category II [1], whereas current LEU facilities are classified as Category III. The key categorization difference between normal LEU production and HALEU production is highlighted in Table 1: “10,000 grams or more of uranium-235 (contained in uranium enriched to 10 percent or more but less than 20 percent in the ^{235}U isotope).” Given these definitions, material enriched between 5 and 10wt% is considered HALEU but falls into a Category III facility.

The difference in category from LEU to HALEU—and the elevation of the nuclear facility from Category III to Category II—fundamentally increases various aspects of the facility’s security posture. However, such elevation is not uniform across all aspects of defense-in-depth. The most significant difference is that HALEU must be stored in vault-type, locked cabinets and/or locked conveyances, whereas vault type or locked conveyances are not mandated by U.S. NRC Title 10 Code of Federal Regulations part 73 (10CFR73 physical protection of plants and materials) for

Category III material storage. The IAEA physical security recommendation for Category II (INFCIRC/225/Revision 5) specifies that Category II material should be stored within the site's protected area, which slightly deviates from the 10CFR73 requirement that Category II materials be stored in a vault-type room or locked cabinet within a controlled access area. Per IAEA, Category III material must be stored in the site's limited access area. The regulatory requirements and international guidance for physical security of Category II special nuclear material apply to HALEU and HEU in the same way.

3 Nuclear Fuel Cycle Assessment

3.1 Enrichment and Downblending

The physical security requirements and capital costs for enrichment at various levels are key to understanding the deployment of a HALEU supply chain. Based on guidance in INFCIRC/225, enrichment above 10wt% must be conducted in a Category II facility, which can require significant capital investments to license, build, secure, and operate [2]. However, over 90% of the separative work required to enrich natural uranium to 20wt% ^{235}U is needed to get to the 10% enrichment range, and enrichments to 10wt% could be conducted in a Category III facility with the same physical security category building as those used to enrich LEU. If a two-tier enrichment system is used to leverage existing LEU enrichment infrastructure at an existing facility, the size and security burden of an additional Category II facility may reduce the costs of HALEU production while meeting existing safeguards and security requirements. The physical security posture at a hybrid facility must be well understood prior to implementation.

The output of enrichment facilities would be transported via UF_6 that are transported in cylinders. Minimizing the handling of UF_6 HALEU cylinders would result in more efficient operations and lower security related fuel fabrication costs. The co-location of HALEU facilities with an LEU enriching facility could decrease the cost and security risks of transportation; specifically, sharing a facility perimeter would reduce security costs. If facilities are not collocated or at least optimally located, transportation risk would be greater, and costs will be higher. These integrated supply strategies are methods to optimize operations and improve security.

In general, uranium enrichment is a concern because the same technology that can produce LEU for reactor fuel can also be used for enrichment of HEU for nuclear weapons. It can be challenging to identify differences between HALEU and HEU containers, process equipment, material batches, etc.; these differences are what inspectors and others look for to identify nefarious activities. With smaller containers required for nuclear safety reasons, HALEU becomes more portable while also containing more U-235 than LEU in the same size container. Material enriched to less than 20wt% but higher than current LEU nuclear reactor fuel, is considered of higher concern from a nuclear security perspective.

Though the United States does not currently have an existing infrastructure for HALEU production via enrichment beyond 5wt%, industry is aggressively pushing to deploy additional enrichment capacity up to 20wt% with the necessary NRC approvals. According to World Nuclear News, Centrus is on track for a HALEU demonstration by the end of 2023. The U.S.-based company has completed construction of a cascade of advanced centrifuges in Piketon, Ohio [3]. The United States has infrastructure for downblending from HEU stocks, but the feedstock for downblending is limited.

Commercial fuel cycle facilities with the intention of creating HALEU will require additional security beyond what is already deployed for LEU only facilities. Transportation of natural or depleted feedstock already has a functional supply chain and increasing the overall demand may impact the transportation supply. However, there are insufficient commercial transportation supplies for HALEU, both in the U.S. and internationally. The U.S.'s National Nuclear Security Administration has a container that is licensed to handle HALEU called an ES-3100, but it was not designed with a HALEU fuel cycle in mind therefore it is not optimized to handle the material.

3.1.1 Enriched Feed Material Packaging and Transportation

Low enriched uranium is typically stored and transported via 2.5-ton and 10-ton UF_6 cylinders. Industry is currently working to enable these cylinders for HALEU use. Minimizing the handling of UF_6 HALEU cylinders would result in more efficient operations and lower security-related fuel fabrication costs for the vendor.

One of the concerns with HALEU fuel is the development of fuel transport containers that can address the expected demand for shipping HALEU material and the subsequent new security risks that arise. If industry's efforts to license the 2.5-ton and 10-ton cylinders for HALEU contents is unsuccessful, much smaller cylinders will need to be used. Replacing the larger 48-inch (Model 48X or 48Y) and 30-inch (Model 30B) with much smaller cylinders in the 5-to-8-inch range (i.e., the same class as Model 8A and 5B) will create an order of magnitude increase in the number of containers and requisite handling operations. The obvious implication of this increase is the need for increased quality management, Nuclear Material Accounting and Control (NMAC), and surveillance of these smaller HALEU UF_6 cylinders. More material moves represent higher potential for theft or material loss during transport. Increasing the number of small cylinders per locked conveyance may be a plausible mitigation to theft; however, for this method to be a viable option, an approved shipping package must be designed and licensed for HALEU.

3.2 Fuel Fabrication

Fuel fabrication facilities were identified as the highest significant security concern in the HALEU nuclear fuel cycle. The security concern is driven by the higher enrichment of the particle and metallic fuels. Both the uranium metal fuel for the reference sodium cooled fast reactor and the TRISO fuel for the reference helium-cooled pebble bed reactor are enriched to 15-20wt% ^{235}U. According to the NRC definitions (Table 1), a fuel fabrication facility for this range of HALEU would be classified as a Category II facility.

Category II HALEU fuel fabrication facilities will require enhanced physical security requirements compared to Category III fuel fabrication facilities. Unalloyed and unprocessed feed material in bulk form early in the fuel fabrication process is the most at-risk material because it is in an attractive form and not diluted by non-nuclear material components. The concern regarding this material decreases as the fuel is processed because additional non-nuclear materials are mixed with the special nuclear material to create the final fuel form. Dilution of the special nuclear material will make protracted theft less probable, a concept applicable to both U metal and TRISO fuel further along the fuel fabrication process. The non-nuclear materials used for HALEU fuel fabrication present new security challenges that could affect sabotage risk (e.g., higher fabricating temperatures and pressures). However, at a HALEU facility, the nuclear material sabotage risk and the safety risk to the public is comparable to the risk at existing LEU facilities.

Most sabotage concerns are related to the industrial hazards associated with the fabrication processes as opposed to any radiological hazards, because the criticality safety design of fuel fabrication equipment should be sufficient to minimize any credible criticality event. In other words, the casting furnace, solgel tank, fluidized bed, etc., are favorably designed such that a critical mass or critical geometry may not be achieved under any practical fabrication facility sabotage scenario. Also, though unirradiated uranium is mildly radioactive, a substantial release of uranium does not pose a greater risk to worker safety, public health, or the environment than the non-radiological (i.e., chemically toxic) consequence of the uranium-carrying chemicals associated with such a release. For example, destruction of a metallic fuel casting furnace would disperse metallic sodium and uranium, which are pyrophoric in air. Also, destruction of internal gelation vats or fluidized beds used for TRISO production can lead to releases of nitric acid, urea, and other various organic compounds considered to be, at minimum, an irritant to the human body. Thus, the nuclear or radiological risk of sabotage for HALEU fuel fabrication can be considered comparable to that for LEU fuel fabrication.

The coating process of TRISO particles involves substantial industrial hazards with flammable and possibly pyrophoric gases as well as custom-designed furnaces that can operate at higher pressures and temperatures. Given the hazard profile associated with the equipment and nature of the process, the equipment could be more susceptible to sabotage. Additionally, if it serves as feed material for the fuel fabrication process, the received UF_6 material could be susceptible to sabotage. It is toxic, and conversion steps generally include heat, chemical reactors (vessels), and strong acids, all of which can be manipulated to destroy equipment or injure personnel.

3.3 Spent Fuel On-Site Storage and Transportation for Disposal

The security assessment for spent fuel from the reference advanced reactors is provided relative to the security of a baseline LWR spent fuel assembly. A security analysis of the two fuel types is performed in the context of both on-site storage of used nuclear fuel and transportation of used nuclear fuel to a waste repository or other destination. The security analysis relies on both the physical condition of the fuel after irradiation and the storage location and access points for the spent nuclear fuel.

The security analysis of spent nuclear fuel storage and transportation focuses on two main design basis threats, which are radiological sabotage, and theft. For radiological sabotage, the key parameters are the quantity and activity of radionuclides that are contained within the spent fuel and the dispersibility of these radionuclides. The radiological dose is used as a representative surrogate for the quantity and activity of the radionuclides present in the spent fuel. The dispersibility of the radionuclides is defined by the physical condition of the spent fuel and is evaluated for each fuel type. For theft or diversion, key parameters are material attractiveness and material access. For material attractiveness, the quantity of plutonium or the remaining HALEU in the spent fuel is used as a representative surrogate. The material access is defined by the physical condition and on-site storage of spent fuel.

To support the analysis of spent fuel storage and transportation, representative characteristics of spent nuclear fuel (SNF) were computed, such as decay heat and dose rate. These characteristics were then used to qualitatively evaluate of storage configurations and transportation configurations.

3.3.1 Spent Fuel Storage and Transport Conditions

The spent TRISO fuel from the reference pebble bed reactor design is stored in air under dry conditions. For the storage of the spent TRISO fuel, similar physical security measures as those currently employed for dry storage of spent LWR fuel are expected to be acceptable. The risks and consequences of both radiological sabotage and theft or diversion of the reference TRISO spent fuel are slightly less than for LWR spent fuel dry storage. Due to the dilution of uranium (and plutonium) in graphite, a great many pebbles would require theft before an appreciable amount of material could be obtained. Also, because of the robust SiC layer of TRISO particles and the very low decay heat energy density of pebbles, air-cooled interim storage of pebbles has innate resilience to sabotage.

Transportation of spent TRISO fuel from the reference pebble bed reactor would also mimic the security profile of existing SNF shipments, with the main difference being an increase in the number and size of shipments. Risks and consequences are of a similar order of magnitude to those of LWR spent fuel.

The spent fuel from the reference sodium cooled fast reactor is initially stored in either a water or liquid sodium pool and later, after sufficient cooling has occurred, in air under dry conditions. For the storage of the spent fuel from the reference sodium cooled reactor, similar physical security measures as those currently employed for spent LWR fuel in the spent fuel pool and under dry storage are expected to be acceptable. The risks and consequences of both radiological sabotage and theft or diversion of the sodium cooled spent fuel are slightly greater than LWR fresh and spent fuel storage in water pools. The reference sodium cooled spent fuel will also be stored in underwater or pool-storage. However, because the spent fuel will have a higher fissile density and burnup (i.e., concentration of radionuclides in spent fuel), the security risk (theft and sabotage) should be considered somewhat greater for the reference sodium cooled spent fuel than for LWR spent fuel.

The reference sodium cooled SNF transport is like that of standard LWR fuel. Metallic fuel from the reference sodium cooled reactor may have increased consequences depending on transport configuration. No modern cask for metallic SFR fuel was identified, so a more hypothetical approach was taken. One would expect that to license a cask, it would have similar margins against failure to existing LWR casks, and therefore one would expect a licensable cask to have similar security requirements to existing LWR fuel.

As discussed in the following subsections, sabotage consequences related to the spent fuel from the reference sodium cooled reactor were slightly higher than for LEU-based spent fuel, but the difference is not significant. The risks and consequences of both radiological sabotage and theft or diversion of spent TRISO fuel are slightly less than for LWR spent fuel.

3.3.2 Source Term Calculation

Nuclear engineering computer codes were used to calculate the quantity of plutonium and radiological dose from the two reference HALEU designs relative to a reference LWR fuel assembly. The plutonium content expected in each type of spent nuclear fuel is used as a surrogate for material attractiveness and the dose rate expected three feet from an irradiated fuel element with no shielding is used as a surrogate for quantity and activity of radionuclides present. The calculational approach uses several well-known nuclear engineering codes in series to establish irradiation(s) conditions, simulate the burnup of the fuel elements, and then calculate the unshielded dose three feet from the fuel element, as illustrated in Figure 1.

The TRITON code generated the nuclear libraries for each new fuel type [4]; this information was provided to ORIGEN to calculate the quantity of each isotope within the fuel. Finally, the MAVERIC code calculated isotope concentrations to calculate the expected unshielded dose at

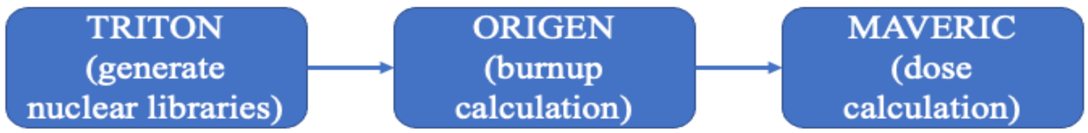

Figure 1. Simulation approach for Pu content and dose rates related to spent fuel.

three feet from the fuel element for each burnup and cooling time [4].

Table 2 shows that the sodium cooled reference reactor has less Pu when compared with the baseline LWR; this is because the reactor starts with less fuel mass than an LWR. However, per mass of initial fuel, the sodium cooled reference reactor is more effective at producing Pu per year than an LWR. Table 3 shows the quantity of Pu in 1000 TRISO pebbles from the reference pebble bed reactor as well as the expected quantity of Pu that is discharged from the reference reactor in a year. It would take about 100,000 pebbles to accumulate a significant quantity of Pu, which is equivalent to the Pu content in about 300 LWR assemblies. The reference pebble bed reactor is much smaller than an LWR fuel assembly and therefore produces less Pu.

Table 2. Plutonium content in reference sodium cooled fast reactor's spent fuel.

	Kg Pu per Assembly	Kg Pu per Year Discharged
Reference sodium cooled reactor 100 GWd/MTU	5.42	143
Reference sodium cooled reactor 75 GWd/MTU	4.34	153
LWR 55 GWd/MTU	6.86	263

Table 3. Plutonium content in reference pebble bed reactor's spent fuel.

	Kg Pu	Kg Pu per year discharged
Reference pebble bed reactor, 165 GWd/MTU	0.091 per 1000 pebbles	4.6
Reference pebble bed reactor, 125 GWd/MTU	0.082 per 1000 pebbles	5.4
Reference pebble bed reactor, 100 GWd/MTU	0.073 per 1000 pebbles	6.1
LWR, 55 GWd/MTU	6.86 per assembly	263

3.3.3 Radiological Sabotage

Table 4 shows the results of the calculation of an unshielded dose at three feet from a reference sodium cooled spent fuel assembly. Although the reference sodium cooled spent fuel burnup is greater than that of the LWR assembly, the sodium cooled assembly dose is less than a LWR assembly. This is in part due to the sodium cooled spent fuel having less fuel mass than an LWR fuel assembly.

Table 5 shows the results of the calculation of unshielded doses at three feet from a TRISO fuel pebble. Each TRISO pebble contains less fissile material, which means less fission products, and therefore the dose from a single TRISO pebble is lower than an LWR assembly. It is unrealistic to expect a single pebble to be stored individually, therefore, in practice, multiple spent TRISO pebbles will be stored together, with thousands in a single container and large numbers of containers in a single location.

Table 4. Dose rate from unshielded spent fuel at three feet (rem/hr) for reference sodium cooled fast reactor.

	Metallic fuel		**Baseline LWR**
Case	**100 GWd/MTU**	**75 GWd/MTU**	**55 GWd/MTU**
5 years decay	1345	983	2560
20 years decay	657	500	782
40 years decay	405	309	470

Table 5. Dose rate from unshielded spent fuel at three feet (rem/hr) for reference pebble bed spent fuel.

	TRISO fuel (1000 pebbles)[1]		**Baseline LWR**
Case	**165 GWd/MTU**	**100 GWd/MTU**	**55 GWd/MTU**
5 years decay	390	204	2560
20 years decay	159	96	782
40 years decay	96	60	470

For radiological sabotage, the key parameters are the quantity and activity of radionuclides that are contained within the spent fuel and the dispersibility of these radionuclides. The radiological dose calculated for the reference sodium cooled fuel, the reference pebble bed fuel, and the reference LWR fuel is used as a representative surrogate for the quantity and activity of the radionuclides present in the spent fuel. For the sodium cooled reactor, the dose is slightly less than for LWR fuel. The dose from a single TRISO pebble is 1,500 to 4,000 times less than for LWR

[1] Dose rates for 1000 TRISO pebbles is derived by multiplying the single dose rate by 300. The factor is 300 versus a direct 1000 increase because the self-shielding of the pebble and location of the individual pebble. For instance, the dose contribution of a pebble in the center of the container versus one on the edge would be low due to other pebbles shielding the gamma/neutrons as well as being further from the measured dose rate location. The factor was based on a dose rate calculation with 1000 pebbles compared to a dose rate calculation with 1 pebble.

fuel. However, typically thousands of pebbles would fit into one of the existing LWR spent fuel canister designs and would have a dose about 300 times greater than a single pebble[2]. For comparison, the dose rate from a loaded TRISO spent fuel canister is 5 to 15 times less than for a LWR fuel assembly.

Regarding dispersibility, significantly more gaseous fission products reside in the fuel cladding gap for metal fuel (such as the reference sodium cooled reactor design) than for oxide fuel such as the baseline LWR fuel [5]. If cladding was breached during an act of sabotage, there would be a greater release of radioactive noble gases, such as xenon and krypton. Because highly radioactive fission gases have very short half-lives, the dose from breached-cladding fuel is dominated by the radioisotopes that remain in the fuel several days after discharge from the reactor. The gaseous release from both the reference sodium cooled fuel and the LWR fuel would have similar dose consequences [6]. The TRISO fuel contains multiple barriers to prevent the release of radioactive materials. In most kinetic events, it is expected that the fuel elements may be damaged, but individual TRISO particles likely would not, thereby significantly reducing the risk of airborne radioactive material [7]. Given the smaller quantity and activity of radionuclides in a TRISO pebble spent fuel canister and the reduction in radionuclide dispersibility, the consequence of radiological sabotage on spent TRISO fuel is expected to be lower than that for LWR spent fuel.

3.3.4 Theft or Diversion

In spent fuel, both enriched uranium and plutonium will be present. Enriched uranium from the initial fresh fuel will remain in the spent fuel, and this material can also be targeted for theft to achieve a significant quantity goal. The Pu contained in the reference sodium cooled reactor spent fuel is approximately the same as for LWR fuel. For material access, the reference sodium cooled spent fuel assemblies are stored in much the same way as for LWR spent fuel assemblies and accessibility is similar regardless of whether they are stored in a water pool or a sodium pool. Therefore, the access is expected to be approximately the same for the reference sodium cooled reactor and an LWR. As mentioned previously, for TRISO fuel, the plutonium contained in a single spent fuel element is about 1,000 times less than for LWR fuel, and about 1,000 fuel elements will fit into a TRISO spent fuel canister. However, the current design concept for the TRISO disposal canisters is much smaller than the LWR containers, and as such, more portable. Therefore, the access to remove a canister is expected to be lower than a LWR container. Extraction of Pu from spent TRISO fuel is very difficult because of the TRISO layered fuel design, making it much less attractive than other fuel types. Each pebble is coated with layers of pyrolytic carbon and silicon carbide which are more resistant to acid dissolution, and the U and Pu in the TRISO fuel are dispersed in many small particles.

Physical security measures for the transportation of spent fuel for both the reference sodium cooled reactor and reference TRISO pebble bed reactor are similar to those currently employed for spent LWR fuel.

4 Conclusion

This study investigated the nuclear security impacts of HALEU relative to LEU for key steps in the nuclear fuel cycle, including enrichment, fuel fabrication and fresh fuel transportation, reactor

[2] This estimate does not account for any limited shielding provided by the canister.

operation, and irradiated fuel storage and transportation. Conclusions for the reference sodium cooled reactor and reference TRISO pebble bed reactor designs are summarized below.

- There is no significant change in the nuclear security risk profile for HALEU relative to LEU when considering sabotage risks and the likelihood of theft; consequences of theft were not evaluated in this study.
- There is a possible increased opportunity for theft or sabotage during transport, stemming from the smaller size of shipping containers and subsequent expected increased frequency of transport of feedstock and fresh fuel, and spent fuel.
- The radiological sabotage consequences for the spent fuel from the reference sodium cooled reactor stored in water pools were slightly higher than for LEU-based spent fuel, but the difference is not significant.
- The radiological sabotage consequences for TRISO spent fuel stored in dry storage were slightly less than for LEU-based spent fuel, but the difference is not significant.

Acknowledgements

The authors would like to express their appreciation to the U.S. DOE NNSA Office of International Nuclear Security for supporting this work.

Works Cited

[1] Nuclear Regulatory Commission, "Appendix M to Part 110—Categorization of Nuclear Material," 17 April 2020. [Online]. Available: https://www.nrc.gov/reading-rm/doc-collections/cfr/part110/part110-appm.html.

[2] International Atomic Energy Agency, "Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5)," International Atomic Energy Agency, Vienna, 2011.

[3] World Nuclear News, "Centrus on track for HALEU demonstration by year-end," 10 February 2023. [Online]. Available: https://www.world-nuclear-news.org/Articles/Centrus-on-track-for-HALEU-demonstration-by-year-e. [Accessed April 2023].

[4] W. A. Wieselquist, R. A. Lefebvre and M. A. Jessee, *SCALE Code System, ORNL/TM-2005/39, Version 6.2.4,* Oak Ridge, TN: Oak Ridge National Laboratory, 2020.

[5] D. Grabaskas, M. Bucknor, J. Jerden and R. Denning, "A Mechanistic Source Term Calculation for a Metal Fuel Sodium Fast Reactor," in *International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development*, Yekaterinburg, Russia Federation, 2017.

[6] C. E. Beyer and P. M. Clifford, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," Pacific Northwest National Laboratory, Richalnd, Washington, 2011.

[7] Nuclear Reulatory Commission, "TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents," Office of Nuclear Regulatory Research NUREG/CR-6844, Vol. 1, Washington, DC.