POTENTIAL CHALLENGES WITH STORAGE OF SPENT (IRRADIATED) ADVANCED REACTOR FUEL TYPES

Prepared for

U.S. Nuclear Regulatory Commission
Contract No. 31310018D0001
Task Order No. 31310018F0113

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August 2019

ABSTRACT

This report discusses potential challenges and issues associated with storage of spent (irradiated) advanced reactor fuel (ARF) based on a review of storage experience with non-light water reactor (LWR) fuel. Primary considerations were given to identifying degradation processes that may challenge canister materials and the configuration of spent fuel during storage, as well as storage designs used for dry storage of ARF types. Non-LWR fuel for which information was reviewed includes solid coated particle fuel, commonly referred to as tristructural isotropic (TRISO), and nuclear metal fuel including uranium alloys such as U-Pu, U-Fs, U-Zr, U-Mo, and U-Pu-Zr, often with Na between fuel and stainless steel cladding, characteristic of compact fast reactors. Design parameters and characteristics of the two ARF types were examined considering key topics in NUREG–2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities.” Challenges identified are discussed within the context of safety evaluation topics including structural, thermal, shielding, criticality, material, and confinement performance, all of which form the design basis of dry storage systems (DSSs) and dry storage facilities (DSFs). Additionally, this report examines potential issues associated with storage experience with existing non-LWR performance history in relation to ARF performance requirements envisioned for next generation nuclear reactors.

For TRISO fuel, the U.S. Nuclear Regulatory Commission (NRC) Site-Specific License No. Special Nuclear Material (SNM)-2504 for the Fort St. Vrain (FSV) Independent Spent Fuel Storage Installation (ISFSI) was issued to store irradiated TRISO-coated particles inside prismatic block fuel elements. With respect to NUREG–2215 review topics, TRISO fuel design parameters in general appear to be within design limits for the DSSs and DSFs approved by NRC. However, TRISO fuel design parameters envisioned for modern high-temperature gas--cooled reactors include a maximum fuel burnup of 150–210 GWd/MTU and uranium enrichment levels up to 20 weight percent U-235, which poses a potential challenge that may require changes to the current regulatory fuel burnup and enrichment limits and related criticality control requirements currently defined for storing LWR SNF. Detailed thermal and materials analyses for TRISO fuel discharged from fluoride salt-cooled high-temperature reactors would be needed to address storage of spent TRISO fuel with higher decay heat and residual salt material.

Metal fuel design consists of a wide variety of metallic compositions, as well as cladding materials, which have been exposed to a variety of irradiation conditions. Possible degradation mechanisms are material-dependent and are functions of environmental (e.g., relative humidity, pH, and chemistry), thermal, mechanical, and irradiation conditions. Based on these degradation mechanisms, potential challenges to storage of ARF are identified. Metal fuel has characteristics important to DSSs and DSFs that differ from light water reactor spent fuel, which may require detailed analysis to fully understand performance for storage of these spent fuel types. These differences include factors unique to the composition of the fuel, thermal limits, criticality, and material performance. Given the characteristics of spent metal fuel and the challenges observed with existing storage system performance, it is expected that certification of DSSs and DSFs for spent metal fuel, or its corresponding converted high-level radioactive waste forms, would need to address special design parameters (different than for light water reactor spent fuel) that would be necessary to support safety assumptions for storage of spent ARF.

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ABBREVIATIONS/ACRONYMS

ARF	advanced reactor fuel
AVR	Arbeitsgemeinschaft Versuchsreaktor
CFR	Title 10 of *Code of Federal Regulations*
CNWRA®	Center for Nuclear Waste Regulatory Analyses
DOE	U.S. Department of Energy
DSF	dry storage facility
DSS	dry storage system
EBR-II	Experimental Breeder Reactor-II
FHR	fluoride salt-cooled high-temperature reactor
FSV	Fort St. Vrain
HFEF	Hot-Fuel-Examination-Facility
HTGR	high-temperature gas-cooled reactor
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
ISFSI	Independent Spent Fuel Storage Installation
LWR	light water reactor
NRC	U.S. Nuclear Regulatory Commission
SNF	spent nuclear fuel
SNM	Special Nuclear Material
TRISO	tristructural isotropic

ACKNOWLEDGMENTS

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

The authors would like to thank Osvaldo Pensado for his technical review and David Pickett for his programmatic review. Review comments by the NRC staff helped improve the report. The authors also thank Arturo Ramos for providing word processing support in preparation of this document.

QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

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

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

1 INTRODUCTION

1.1 Background

As the U.S. Nuclear Regulatory Commission (NRC) staff prepares for regulatory interactions and potential license applications for non-light water reactor (LWR) technologies, there is a need to develop an understanding of the potential challenges associated with regulating the long-term storage, transportation, and disposal of advanced reactor fuel (ARF) types. Revisions may be needed to guidance documents and rules promulgated in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71 and 10 CFR Part 72 related to spent ARF types. Potential ARF types that may be subject to NRC regulation in the future include metallic fuels, uranium fuels for high-temperature gas-cooled reactors (HTGR), and molten fuel salt.

The Center for Nuclear Waste Regulatory Analyses (CNWRA®) has been tasked with identifying and assessing the significance of potential technical challenges and issues associated with the storage, transportation, and disposal of ARF types. This report assesses possible technical issues that may need to be addressed in safety reviews of storage facilities and cask systems associated with storage of spent ARF types. Primary considerations were given to identify degradation processes that may challenge current canister materials and storage designs used for dry storage of ARF types. NonLWR fuel for which information was reviewed includes solid coated particle fuel, commonly referred to as tristructural isotropic (TRISO), and nuclear metal fuel (uranium alloys such as U-Pu, U-Fs, U-Zr, U-Mo, and U-Pu-Zr, often with Na between fuel and stainless steel cladding) of compact fast reactors.

1.2 Purpose and Scope

The third report in this series (Hall et al., 2019a) reviewed available operating experience with storage of spent (irradiated) ARF types. This fourth report identifies potential challenges for storage of spent ARF types by considering known or expected characteristics of the ARF and characteristics of storage facility designs in the context of applicable NRC safety review topics for dry storage of spent nuclear fuel (SNF). Additionally, this report reviews characteristics of spent ARF that could be important for NRC during a safety review for dry storage systems (DSSs) or dry storage facilities (DSFs). Design parameters and characteristics of the two ARF types were examined within the context of safety review topics applicable to dry storage of spent fuel as identified in NUREG–2215. Potential challenges associated with storage experience, including any degradation mechanisms noted, are identified based on existing non-LWR fuel performance data. For this report, challenges identified are discussed in the context of relevant safety evaluation topics, including structural, thermal, shielding, criticality, material, and confinement performance, all of which form the design basis of DSSs and DSFs.

2 CHARACTERISTICS OF ARF TYPES IMPORTANT TO DRY STORAGE

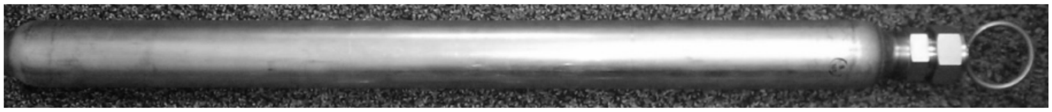
NRC regulations in 10 CFR Part 72 establish requirements for licensing DSSs and DSFs. For storage of spent fuel, the characteristics of the fuel that need to be evaluated in NRC safety reviews include type, number of spent fuel assemblies, maximum and minimum initial enrichment of the fuel, burnup, minimum cooling time of the spent fuel in storage pool, maximum decay heat, maximum mass of the spent fuel, radioactivity level, and condition of the spent fuel and cladding.

In an earlier report from this project (Hall et al., 2019a), the characteristics and storage experience of spent TRISO fuel and metal fuel were discussed. The NRC Site-Specific License No. Special Nuclear Material (SNM)-2504 for the Fort St. Vrain (FSV) Independent Spent Fuel Storage Installation (ISFSI) was issued to store irradiated TRISO-coated particles inside prismatic block HTGR fuel elements (NRC, 2011). The performance requirements of TRISO-coated particle fuel have been established for modern HTGRs (NEA, 2014; INL, 2010). The key design parameters and characteristics affecting storage of spent TRISO fuel, as well as LWR SNF, and the types of DSSs are listed in Table 2-1. The design limits of the FSV ISFSI are based on the thermal and radiological characteristics of the FSV spent fuel elements (DOE, 2010).

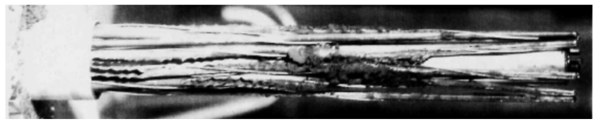
Hall et al. (2019a) explained that spent nuclear metal fuels from fast reactors have been stored in containers in wet and dry conditions at Idaho National Laboratory (INL). These storage systems are DOE-certified, but not NRC-certified (NWTRB, 2017). Figure 2-1(a) is a photo of the Type 304 stainless steel container used to store the metal SNF at Idaho Nuclear Technology and Engineering Center (INTEC) at INL (Pahl, 2000; Pahl et al., 1996). Each container, which is about 5 cm [2 in] in diameter and 78 cm [30 in] in length, contains up to 12 fuel pins (approximately 0.7 kg of highly enriched U). The container was closed by leak-tight caps. Ambient air was originally used as backfill gas in most of the containers, but dry argon was used later to create an inert environment. In the mid-1990s, water from the storage pool was found to have leaked into some containers. It was speculated that the leakage was most likely due to improperly tightened lids. The chlorine ion-containing water (chlorine in the range of 50–350 ppm) was speculated to have compromised the degraded cladding from reactor operation leading to cracking of the cladding and water reaction with Na and fuel (DOE, 2012). Figure 2-1(b) shows one example of the cracked cladding and degraded fuel from one container. It was observed that U reaction with O_2 and H_2O produced fuel oxide particulates and uranium hydrides and Na reaction with water produced H_2 and NaOH.

The dry storage container for spent metal fuel in the Radioactive Scrap and Waste Facility at INL has carbon steel as an inner container and stainless steel as an outer container. The inner container is backfilled with argon and closed with bolted and gasketed lid, but the outer container is closed with a weld. The fuel-loaded, double-encapsulated container is then set in a dry, below-grade carbon steel liner, which is approximately 2 m [6.5 ft] below the soil surface. The liners are cathodically protected from corrosion and have shield plugs at the tops to shield radiation and prevent water intrusion. The top 10 cm [3.9 in] of the liner is above the ground surface to further prevent water entry. The ground around the liner also provides shielding for the radiation fields associated with the spent fuel. The thermal limit for the driver fuel is 300 W for two subassemblies stored in one liner (with each subassembly containing about 31 fuel pins) and the thermal limit for the blanket subassemblies is 180 W for 6 subassemblies in one liner (with each subassembly containing about 19 fuel pins) (Clarksean and Zahn, 1995). Clarksean and Zahn (1995) indicated that these thermal limits were selected to ensure that the maximum temperature during storage at the facility would not exceed 400 °C [752 °F] and 370 °C [698 °F] for spent driver and blanket fuels, respectively, to avoid rupture of cladding by internal stress at high temperature. In this configuration, moisture was also found to penetrate through the four barriers (i.e., the carbon steel liner, outer container, inner container, and cladding) resulting in cladding rupture and fuel degradation forming uranium oxides and hydrides (DOE, 2012). The damaged fuel pins from wet and dry storage conditions were placed into fuel containers as the one in Figure 2-1(a). Eight containers were placed into a transfer basket as shown in Figure 2-2. Two fully loaded baskets were then placed into a storage can. The can was placed into a Hot-Fuel-Examination-Facility (HFEF)-5 cask (DOE, 2012). The HFEF-5 cask was then loaded into the dry storage carbon steel liner at INL.

Table 2-1. Design parameters and characteristics of light water reactor and TRISO-coated particle fuel and storage systems			
Parameter	**LWR Fuel**	**FSV Fuel (DOE, 2010)**	**Coated Particle Fuel Service Conditions Proposed for HTGR**
Fuel Type and Component	UO_2	TRISO-coated $ThUC_2$ particles in prismatic block fuel elements	TRISO-coated particles in pebble style or prismatic block fuel elements
Enrichment (weight percent U-235)	Maximum initial enrichment of 5 percent (NUREG–2215)	Maximum initial enrichment of 93.15 percent	Low-enriched TRISO fuel with an initial enrichment less than 20 percent (INL, 2010)
Burnup (GWd/MTU)	Maximum burnup of 60 GWd/MTU (NUREG–2215)	Maximum burnup of 52 GWd/MTU	150-200 GWd/MTU (NEA, 2014)
Thermal Characteristics	Maximum cladding temperature of 400 °C [752 °F] for normal conditions (NUREG–2215)	Maximum fuel temperature of 399 °C [750 °F]; maximum and average decay heat of 150 W and 85 W per fuel element	Not part of the proposed fuel service conditions
Radiological Characteristics	Storage container: maximum neutron fluence of 2.63×10^{16} n/cm^2 (NUREG–2215)	Storage container: maximum gamma and neutron fluence of 4×10^{11} rad and 5×10^{14} n/cm^2 Fuel: maximum gamma and neutron radiation of 2.97×10^{14} photons/s and 3.31×10^5 n/s	Not part of the proposed fuel service conditions
Storage System Type	Canister-based or direct-load metal cask dry storage systems	Direct-load metal cask dry storage system	Not part of the proposed fuel service conditions



(a)



(b)

Figure 2-1. (a) Stainless steel containers about 5 cm [2 in] diameter and 78 cm [30 in] long for containing spent metal fuel elements in storage in water pool (DOE, 2012) (b) cracked cladding and degraded fuel from some storage containers with water leaked in (Pahl, 2000; Pahl et al., 1996)

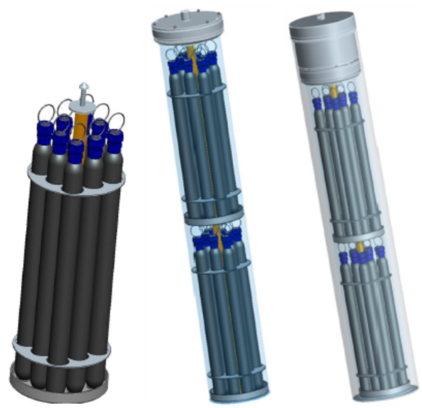


Figure 2-2. Storage of damaged spent metal fuels. Left image: a fuel bottle transfer basket loaded with 8 fuel bottle containers; middle image: blanket storage can loaded with two baskets; right image: a hot-fuel-examination-facility-5 cask containing the loaded storage can (DOE, 2012)

Table 2-2 lists the characteristics of spent metal fuel affecting SNF storage and the DOE-certified wet and DSSs along with those for the LWR SNF for comparison. As shown in Table 2-2, there is a wide variety of metal fuels and cladding materials that have been exposed to many different irradiation conditions. Furthermore, these fuels are experiencing complicated storage conditions including (i) wet storage, (ii) dry storage, (iii) wet storage followed by dry storage, and (iv) dry storage of the converted ceramic and metallic HLW forms from chemically treating the spent metal fuel. These spent fuels, particularly the cladding, are more damaged than commercial LWR SNFs and the storage experience at INL indicates that they are more susceptible to degradation during storage. In addition, there is limited experience and data on cladding degradation mechanisms such as fuel-cladding chemical interaction. Sodium in the spent metal fuel is pyrophoric and chemically reactive. Therefore, the U.S. Department of Energy (DOE) is treating the spent metal fuel to convert reactive metallic sodium to non-reactive ionic sodium to make it acceptable for future disposal.

Table 2-2. Design parameters and characteristics of light water reactor and metal fuel and storage systems

Parameter	LWR Fuel*	Metal fuel
Fuel type and component	UO_2	U, U-Pu, U-Fs†, U-Zr, U-Mo, U- Pr-Zr (FRWG, 2018)
Enrichment (weight percent U-235)	Maximum 5 percent (NUREG–2215)	26–93 (FRWG, 2018)
Cladding material	Variations of zirconium-based alloys: Zircaloy-4, Zircaloy-2, Zirlo, M5, and others	Variations of stainless steel: 304L, 316, D9, HT9 (FRWG, 2018)
Burnup	Maximum 60 GWd/MTU (NUREG–2215)	0.3–19.8 atomic percent of heavy metal; 38–143 GWd/MTHM (FRWG, 2018)
Thermal Characteristics	Maximum cladding temperature of 400 °C [752 °F] for normal conditions (NUREG–2215)	The thermal power for each driver and blanket subassembly is 150 W and 30 W, respectively. Two driver subassemblies stored in one container results in 220 °C (Clarksean and Zahn, 1995).
Radiological Characteristics	Storage container: maximum neutron fluence of 2.63×10^{16} n/cm^2 (NUREG–2215)	0 to 10,000 R/hr (Hill and Fillmore, 2005)
Storage System Type	Canister-based or direct-load metal cask dry storage systems	Interim wet and dry storage containers

*In this report, the analyses are limited to zirconium-based alloy-cladded UO_2 SNF from domestic commercial LWRs (5 wt% U^{235} maximum enrichment)
†Fissium (Fs) is an alloy left by the reprocessing cycle from EBR-II operation containing 2.4 weight percent Mo, 1.9 weight percent Ru, 0.3 weight percent Rh, 0.2 weight percent Pd, 0.1 weight percent Zr, and 0.1 weight percent Nb.

3 ASSESSMENT OF SPENT FUEL STORAGE

The design parameters and characteristics of the two ARF types identified in Tables 2-1 and 2-2 were evaluated within the context of safety review topics applicable to dry storage of spent fuel identified in NUREG–2215 (NRC, 2017). Because of present uncertainties regarding specific designs of ARFs that might be submitted to NRC in future license applications, this evaluation focused on potential issues for storage of ARFs based on existing experience with similar fuel materials, as well as the fuel service conditions and performance requirements envisioned for the next generation nuclear reactors. Therefore, this chapter evaluates the selected ARF design parameters and characteristics within the context of the safety evaluation topics related to structural, thermal, shielding, criticality, material, and confinement performance.

3.1 Structural Evaluation

As described in NUREG–2215 (NRC, 2017), safe storage requires that the structural integrity of the structures, systems, and components (SSCs) used for dry storage be appropriately maintained under all credible loads and their combinations for normal, off-normal, and accident conditions and natural phenomena effects.

The FSV ISFSI is licensed by NRC to store spent TRISO fuel. The design bases of the FSV ISFSI ensure that the structural integrity of the SSCs is maintained (NRC, 2011). Based on the structural evaluation of the FSV ISFSI, dry storage of spent TRISO fuel using existing NRC-approved DSSs is expected to provide adequate structural integrity for storing spent solid coated particle fuel.

The physical dimensions of metal fuel pins are much smaller than LWR fuel rods and the configurations of the fuel assemblies also differ. Although the canister-based or direct-load metal cask DSSs currently designed for LWR SNF are sufficiently large to hold multiple metal fuel elements and there are many variations of these systems, the internal structures and configurations may need to be modified to hold the metal fuel assemblies. Additionally, differences in thermal and radiological characteristics between spent metal fuel and LWR SNF (e.g., decay heat, discharge burnup, initial enrichment level, and cooling time before storage) may require structural design changes to DSSs, including modifications to the basket structure and to the dimensions, type, location, and configuration of the neutron absorber materials. For example, the internal structure and the loaded content affect the thermal profile inside and outside of the storage system, and the internal configuration of the neutron absorber materials affect the likelihood of criticality. Designing or modifying the design is not expected to be challenging from an engineering perspective, considering the experience gained over the years in managing systems storing LWR SNFs with different characteristics. Structural evaluations for spent metal fuel DSSs and DSFs would be performed as part of the certification process prior to storage of either the spent metal fuel or the converted HLW forms to ensure the structural integrity of SSCs.

3.2 Thermal Evaluation

As described in NUREG–2215 (NRC, 2017), safe storage under normal, off-normal, and accident conditions requires that storage container and fuel material temperatures remain low enough such that the fuel cladding will not be subject to thermal degradation that could lead to gross rupture.

For the FSV ISFSI, the maximum temperature of the spent HTGR fuel elements is 399 °C [750 °F] based on the highest calculated decay heat of 150 W per spent fuel element (NRC, 2011; DOE, 2010). The maximum TRISO fuel temperature falls within the cladding temperature limit of 400 °C [752 °F] for normal conditions of spent LWR fuel storage (NRC, 2017). The fluoride salt-cooled high-temperature reactor (FHR) core power density is four to ten times higher than in an HTGR because the salt coolant provides more efficient cooling of the TRISO fuel. As a result, FHR SNF is expected to have much higher decay heat than HTGR SNF (Forsberg and Peterson, 2015). Therefore, thermal evaluations for dry storage of spent TRISO fuel with higher decay heat would be done as part of the certification process to ensure safe storage of TRISO fuel. However, the FSV spent fuel element has a lower decay heat load by a factor of five to ten compared to LWR SNF (Oak Ridge National Laboratory, 1992). It is expected that existing NRC-approved DSS designs would be capable of storing spent TRISO fuel with higher decay heat.

Temperature during storage of spent metal fuel is one important factor to be considered to ensure cladding integrity. Clarksean and Zahn (1995) selected 400 °C [752 °F] and 370 °C [698 °F] as the temperature limits for EBR-II spent driver and blanket metal fuels, respectively, to ensure a low probability of cladding failure. Guenther et al. (1994) indicated that cladding could rupture at temperatures greater than 400 °C [752 °F] induced by internal stress. Additionally, sensitization (i.e., carbide precipitation at grain boundaries) of cladding is possible at temperatures greater than 200 °C [392 °F], and the temperature limit for spent metal fuel during drying operation needs to be less than 200 °C [392 °F] because of concerns over uranium oxidation and pyrophoric reactions involving uranium hydrides. As discussed in a previous report from this project (Hall et al., 2019b), the end plug for the cladding is welded on to close the fuel pin, which results in many welds. At temperatures of 300 to 400 °C [572 to 752 °F], austenitic stainless steel welds that contain ferrite exhibit a spinodal decomposition of the ferrite phase into ferrite-rich and chromium-rich phases (Alexander and Nanstand, 1995; Chandra et al., 2012). This may lead to weld embrittlement (reduction in fracture toughness) depending on the amount, morphology, and distribution of the ferrite phase in the weld, the composition of the stainless steel, and the time spent in the temperature region.

As mentioned in Hall et al. (2019a), during operation of fast reactors, the cladding is subject to fuel-cladding chemical and mechanical interactions, with the extent depending on the fuel and cladding characteristics and reactor operating conditions. In the fast reactor operating temperature range, stainless steel cladding is also subject to sensitization (Guenther et al., 1996). These degradation mechanisms during reactor operation lead to different initial conditions of the cladding for the storage stage. As a result, thermal evaluations that consider the initial conditions of the cladding, thermal-induced degradation mechanisms for spent metal fuel, and the amount and thermal characteristics of spent metal fuel to be loaded in a DSS would be performed as part of the certification process prior to storage. Thermal evaluations would also be performed as part of the certification process to understand similar respective performance characteristics for the possible HLW forms discussed above, based on the physical and chemical characteristics of these forms.

3.3 Shielding Evaluation

For spent fuel DSSs and DSFs, the design features relied on for shielding must provide adequate protection against direct radiation from the contents. The shielding features should limit the direct radiation dose to the operating staff and members of the public during design-basis normal operating, off-normal, and accident conditions.

For the FSV ISFSI, the maximum gamma and neutron fluence of the fuel storage containers after 50 years are 4×10^{11} rad and 5×10^{14} n/cm^2, respectively (NRC, 2011; DOE, 2010). The maximum neutron fluence on the TRISO fuel storage containers is below the level of 2.63×10^{16} n/cm^2 calculated for dry storage of pressurized water reactor fuel assemblies (NRC, 2017). Based on the shielding evaluation of the FSV ISFSI, existing NRC-approved DSSs is expected to provide adequate shielding for dry-storing spent solid coated particle fuel.

The DOE-certified DSF currently in use at INL for storing spent metal fuel relies on certain features, including the container itself and the below-grade position, for radiation shielding. The actual radioactivity level of the DSS will vary depending on the number of assemblies and the spent fuel characteristics such as fuel type, initial enrichment level, discharge burnup, and inventory of radionuclides. Shielding evaluations would be performed as part of the certification process for DSS and DSF designs for spent metal fuel or the converted HLW forms.

3.4 Criticality Evaluation

For spent fuel DSSs and DSFs, the SSCs must be designed to ensure that the spent fuel remains subcritical under normal, off-normal, and accident conditions involving handling, packaging, transfer, and storage.

The FSV ISFSI license lists the approved contents as irradiated TRISO-coated $ThUC_2$ fuel particles enriched to not greater than 93.15 weight percent U-235 (NRC, 2011). The maximum burnup of the spent TRISO fuel stored at the FSV ISFSI is 58 GWd/MTU (DOE, 2010). The TRISO fuel design parameters envisioned for modern HTGRs include a maximum fuel burnup of 150–210 GWd/MTU (NEA, 2014). The average burnup of the TRISO fuel expected to be discharged from the Mark-1 FHR design is estimated to be 180 GWd/MTHM (Andreades et al., 2014). In addition, the fuel kernel in the international consensus TRISO-coated particle design consists of high-density, low enriched UO_2 or UCO with a uranium enrichment less than 20 weight percent U-235 (INL, 2010). As such, storage of high burnup and enriched TRISO fuel poses a potential challenge that may require changes to the current licensed fuel burnup and enrichment limits for storing LWR SNF (i.e., 5.0 weight percent U-235 enrichment and 60 GWd/MTU burnup in NUREG–2215). Criticality evaluations for storage of TRISO-coated particle fuel with higher burnup and higher enrichment combinations would be performed as part of the certification process to ensure subcritical margins are maintained.

As shown in Table 2-2, metal fuel has much higher U-235 enrichment levels than LWR fuel and the burnup level can also be higher. These higher levels pose a potential challenge that may require changes to the current licensed fuel burnup and enrichment limits. Criticality evaluations for spent metal fuel DSSs and DSFs would be performed as part of the certification process prior to storage of the spent metal fuel or the converted HLW forms.

3.5 Materials Degradation Mechanism Evaluation

The materials performance of storage system SSCs must be adequate under all credible loads and environments for normal, off-normal, and accident conditions such that the spent fuel remains in the emplaced configuration and will not pose operational problems with respect to its removal from storage.

Because the FSV ISFSI is licensed to store spent TRISO fuel discharged from an HTGR, that design is expected to protect against materials degradation for similar TRISO fuel types. The TRISO fuel discharged from FHRs, on the other hand, may contain residual salt coolant. As the temperature decreases, radiolysis of solid fluoride salts in radiation fields will generate fluorine gas that is toxic and potentially corrosive (Forsberg and Peterson, 2015). The fluorine gas can attack the TRISO fuel and SSCs of the storage system. Therefore, materials evaluation for dry storage of spent TRISO fuel with residual salt material would be performed as part of the certification process to ensure that the DSS provides adequate materials performance for storing spent solid coated particle fuel.

For spent metal fuels, including the stainless steel cladding and metal fuel, the degradation mechanisms and extent of materials degradation depend on the environmental, thermal, mechanical, and radiological conditions in the confinement vessel of the storage systems.
For the wet and dry systems used for storage of spent ARF types described in Section 2 and Hall et al. (2019a), the containment environment was sealed from external water or air exposure without filling the internal environment with helium or another inert gas. O_2 and small amounts of moisture that are conducive to corrosion exist in the system. Furthermore, Section 2

indicates that both of the wet and dry storage systems experienced leakage during storage. Considering this storage experience, there are four scenarios for the internal environment that affect the occurrence and the effects of spent fuel degradation mechanisms:

(i). Inert environment without any moisture
(ii). Air environment without any moisture
(iii). Inert environment with some moisture
(iv). Air environment with some moisture

For scenarios (i) and (ii) without any moisture, corrosion processes driven by water do not occur, but some degradation mechanisms driven by thermal and radiological factor—such as low temperature sensitization, thermal aging, and radiation embrittlement—could occur depending on the specific conditions. The stainless steel cladding sensitized during reactor operation can continue experiencing sensitization during storage at temperatures of 200–500 °C [392–932 °F] (Guenther et al., 1996). This low-temperature sensitization mechanism may or may not occur, depending on the storage temperature. The microstructures of most stainless steels can change, given sufficient time at elevated temperatures, and these thermal aging effects may alter the material strength and fracture toughness. As mentioned in Section 3.2, at temperatures of 300 to 400 °C [572 to 752 °F], austenitic stainless steel welds that contain ferrite exhibit a spinodal decomposition of the ferrite phase into ferrite-rich and chromium-rich phases (Alexander and Nanstand, 1995; Chandra et al., 2012), leading to weld embrittlement (reduction in fracture toughness). The amount, morphology, and distribution of the ferrite phase in the weld, the composition of the stainless steel, and the time spent in the temperature region are needed to assess the occurrence and extent of thermal aging. Depending on the neutron fluence, radiation can cause changes in stainless steel mechanical properties, such as loss of ductility, fracture toughness, and resistance to cracking. The effect of radiation embrittlement depends on the initial condition of the cladding, the neutron fluence level during storage, and the duration of storage.

For scenarios (iii) and (iv) (residual moisture scenarios), H_2O can radiolytically decompose into H_2 and oxidizing species, such as H_2O_2, which can chemically decompose into O_2. In moist air, nitrogen oxides (NO_x) and, consequently, nitric acid are also possible radiolytic products. As such, both scenarios (iii) and (iv) contain radiolytic products that can subsequently act as reductants or oxidants to degrade the material in the storage systems. Except for the same degradation mechanisms driven by thermal and radiological factors as described under scenarios (i) and (ii), corrosion mechanisms for stainless steel cladding such as localized corrosion, galvanic corrosion, and stress corrosion cracking could occur depending on the specific conditions. They are discussed briefly as follows.

a. Localized corrosion for passive metals such as stainless steel would initiate when the corrosion potential is greater than the repassivation potential. Jung et al. (2013) calculated corrosion and repassivation potentials for stainless steel in 1 and 5 weight percent H_2O_2 aqueous solutions saturated with oxygen at 25, 75, and 125 °C [77, 167, and 257 °F]. These computations suggest that localized corrosion of stainless steel is not likely in a storage environment with residual moisture at temperatures where aqueous conditions may be established. However, if the moisture contains chloride (e.g., from spent fuel pool water), localized corrosion is likely. Reaction of residual sodium adhering to the cladding outer surface with air and moisture to form caustic Na_2O and NaOH can also induce localized corrosion (Guenther et al., 1996). Information is needed on the chloride concentration, the amount of moisture, the composition and quantity of radiolytic products, and temperature to assess the extent of localized corrosion.

b. As described in Section 2 and Hall et al. (2019a), the inner compartment is usually made from stainless steel. Because there is no galvanic couple, galvanic corrosion between stainless steel cladding and the containment vessel is not likely. If different materials, such as aluminum, are used for internal components to hold the fuel assembly, galvanic coupling is likely. If moisture is present, galvanic corrosion is likely. Information is needed on the amount and chemical composition of moisture, the potential difference of the galvanic couple, and temperature to assess the extent of galvanic corrosion.

c. Stainless steel cladding sensitized during reactor operation can be susceptible to intergranular attack or intergranular stress corrosion cracking (SCC) if stress is present. Chloride is known to induce intergranular SCC of stainless steel. The welded regions of the cladding can be especially susceptible to cracking due to tensile stress in the absence of post-weld stress relief. One particular type of SCC observed for stainless steel cladding is “hot cell rot,” which is speculated to be caused by caustic Na_2O and NaOH formed from residual sodium adhering to the cladding outer surface, and reaction with air and moisture (Guenther et al., 1996). Another type of SCC is irradiation assisted chemical element segregation. SCC can occur quickly, leading to cladding rupture. Information is needed on the initial condition of the cladding, the temperature and radiation level during storage, and the amount and chemical composition of moisture to assess the extent of degradation from SCC.

For all of the scenarios, regardless of whether there is moisture present, if the stainless steel cladding is intact, the spent metal fuel will not experience degradation during storage. However, if the cladding has defects that allow moisture to contact the metal fuel, as observed at INL during wet and dry storage, moisture and O_2 are expected to react quickly with sodium, producing Na_2O, NaOH, and H_2 as in the following reactions

$$4Na + O_2 \rightarrow 2Na_2O$$

$$2Na + 2H_2O \rightarrow 2NaOH + H_2$$

Some of the moisture and O_2 also may react with U metal, forming uranium oxides (UO_x) and pyrophoric hydrides (UH_y), as in the following reactions

$$U + \frac{x}{2}O_2 \rightarrow UO_x$$

$$(1 + \frac{2x}{y})U + xH_2O \rightarrow UO_x + \frac{2x}{y}UH_y$$

Pyrophoric events have occurred in the past when storage cans were opened in air and special care was recommended in handling the fuel (Guenther et al., 1996). The reactions could lead to fuel fracturing and restructuring-swelling. The extent of degradation depends on the presence of sodium. Without sodium sealed in the cladding, fracturing and restructuring-swelling are not expected to lead to extensive damage to the fuel during storage. Information is needed on the extent of cladding breach, the initial condition of fuel, and the amount of moisture in the environment to assess the extent of spent fuel degradation.

Table 3-1 summarizes the feasible degradation mechanisms as discussed above for spent metal fuel and cladding induced by environmental, thermal, mechanical, and irradiation factors.

In addition to cladding and SNF in the storage system as discussed above, there are likely to be other subcomponents (e.g., fuel assembly hardware, basket, and neutron absorbers) that could involve a wide range of materials. Each material can have different degradation mechanisms and the rates of each degradation mechanism are material-dependent and are functions of environmental, thermal, mechanical, and irradiation conditions such as chemistry (e.g., sodium), relative humidity, temperature, and radiation level. Materials evaluations for spent metal fuel DSSs and DSFs certification would be performed prior to storage of the spent metal fuel or the converted HLW forms.

Table 3-1. Environmental, thermal, mechanical, and irradiation-induced degradation mechanisms for cladding and spent metal fuel

Materials	Inert without moisture	Air without moisture	Inert or air with moisture
Stainless Steel Cladding	Low temperature sensitization Thermal aging Radiation embrittlement	Low temperature sensitization Thermal aging Radiation embrittlement	Pitting and crevice corrosion Galvanic corrosion Low temperature sensitization Intergranular attack Stress corrosion cracking (intergranular, “hot cell rot”, irradiation-assisted) Radiation embrittlement Thermal aging
Spent Nuclear Fuel	None	Oxidation Fragmentation Restructuring-swelling	Oxidation Hydriding Fragmentation Restructuring-swelling

3.6 Confinement Evaluation

For spent fuel DSSs and DSFs, the confinement features and monitoring capabilities must be sufficient to limit radiological releases to the environment under normal, off-normal, and accident conditions.

The FSV ISFSI uses carbon steel fuel storage containers in a direct-load metal cask design housed within the Modular Vault Dry Store system. Periodic monitoring of radiological conditions is performed in accessible areas of the ISFSI to ensure radiological posting thresholds are not exceeded. Radiological monitoring results have never indicated degradation of SSCs that shield and confine radioactive material (NRC, 2011). Based on the confinement evaluation of the FSV ISFSI, dry storage of spent TRISO fuel using existing NRC-approved DSSs is expected to provide adequate confinement for storing spent solid coated particle fuel.

At Arbeitsgemeinschaft Versuchsreaktor (AVR), spent spherical pebble style fuel elements with TRISO-coated particles embedded in a graphite matrix are stored at the interim storage facility using the CASTOR-THTR/AVR casks that are made of nodular cast iron and in a direct-load metal cask design (IAEA, 2012, 2010). Each CASTOR-THTR/AVR cask contains two stainless steel dry storage canisters with a capacity of 950 pebbles. Krumbach et al. (2004) reported that some fuel elements were found wet due to leaky sealing of the AVR cans used during the preceding storage in the water pool. The wet fuel elements were stored in separate dry storage canisters that were filled with helium and then sealed with a leak-tight welding. A helium leakage test was conducted to ensure leak tightness after welding, and avoid release of gaseous radionuclides caused by radiolysis of absorbed water in wet fuel elements. Since leak rates from the CASTOR casks were low, release of radionuclides to the environment was concluded to be negligible (IAEA, 2010). Therefore, dry storage of spent TRISO-coated particle fuel using existing NRC-approved DSSs is expected to provide adequate confinement.

As discussed in Section 2, the DOE-certified wet and DSSs for spent metal fuel at INL experienced confinement failure in the past, in spite of the fact that the dry storage system was designed with multiple barriers to prevent radionuclide release. The extent of material degradation caused by the confinement failure was not negligible. These failures show that confinement barriers can be vulnerable and can be challenging to ensure the storage systems meet the confinement requirements. Given the importance of confinement on safety and the extreme reactivity of sodium in SNF with moisture, the ability to measure and monitor compositions of the backfill gas, the conditions of cladding and SNF inside containers, the external surfaces of containers, and the storage facility itself during storage is important in designing, developing, and deploying storage systems for spent metal fuel. Overall, confinement evaluations including monitoring systems for spent metal fuel DSSs and DSFs will be performed as part of the certification process prior to storage of the spent metal fuel or the converted HLW forms. It is expected that it will be feasible to deploy these technologies.

4 SUMMARY AND CONCLUSIONS

Potential challenges with storage of spent ARF include those that affect both canister performance and the configuration of the spent fuel during storage. Storage experience with solid coated particle fuel, commonly referred to as tristructural isotropic (TRISO), and nuclear metal fuel (uranium alloys such as U-Pu, U-Fs, U-Zr, U-Mo, and U-Pu-Zr, often with Na between fuel and stainless steel cladding) typical of compact fast reactors was reviewed to identify relevant degradation mechanisms and possible challenges associated with storage of spent ARF types. An assessment of storage experience revealed some potential challenges with material performance, including canister and fuel degradation, in relation to characteristics of the environmental, thermal, mechanical, and radiological conditions in the confinement vessel of the storage system.

High burnup and enriched TRISO-coated particle fuel poses potential challenges to safe storage beyond those associated with LWR SNF storage. Storage of TRISO fuel discharged from HTGRs with higher burnup and higher enrichment combinations would raise the question of whether subcritical margins are maintained with usage of existing NRC-approved DSSs. Additionally, TRISO fuel discharged from FHR may contain some solid fluoride salts, which can undergo radiolysis at low temperatures and generate fluorine gas that is toxic and potentially corrosive. Degradation mechanisms involving the interaction of solid fluorine salt and TRISO fuel did not appear to be well documented in existing research. Therefore, materials evaluations for dry storage of spent TRISO fuel with residual salt material would need to ensure that a DSS provides adequate materials performance for safely storing spent solid coated particle fuel.

For sodium-bonded fuel with stainless-steel cladding, a variety of storage configurations have been utilized. This type of spent fuel was typically first placed in wet storage followed by dry storage, and in some cases the fuel was chemically treated to deactivate sodium and converted to ceramic or metallic HLW forms for long term storage. In some cases the spent fuel was stored in dry conditions without treatment. At INL, spent sodium-bonded metal fuel degradation occurred while the fuel was in a dry storage container environment due to moisture penetration through each of the four barriers (i.e., the carbon steel liner, outer container, inner container, and cladding), resulting in formation of uranium oxides and hydrides. The importance of maintaining canister integrity for proper confinement of spent nuclear metal fuel was challenged at INL because of interactions of the sodium with the storage environment. Given the possible preexistence of fuel cladding degradation from reactor operation, and the difficulty with monitoring the storage environment, factors affecting storage of sodium-bonded metal fuel have not been fully addressed through past experience. Additionally, differences in thermal and radiological characteristics between spent metal fuel and LWR SNF (e.g., decay heat, discharge burnup, initial enrichment level, and cooling time before storage) may impact structural characteristics of DSSs, such as internal fuel basket dimensions, type, location, and configuration of the neutron absorber materials. Internal structures and the radiological characteristics of the loaded fuel can affect the thermal profile inside and outside of the storage system, and the internal configuration of the neutron absorber materials affect the likelihood of criticality. Under dry storage configurations, cladding degradation could occur at elevated storage temperatures, typically those greater than 400 °C [752 °F], primarily induced by internal stress. Additional thermal-induced degradation mechanisms under dry storage conditions can challenge material performance. Sensitization (i.e., carbide precipitation at grain boundaries) of cladding can occur at temperatures greater than 200 °C [392 °F], as well as uranium oxidation and pyrophoric reactions involving uranium hydrides. The degrees to which these degradation mechanisms affect material performance vary and are affected by the presence of moisture. Degradation from embrittlement (reduction in fracture toughness) is of concern, depending on the amount, morphology, and distribution of the ferrite phase in the weld, the composition of the stainless steel cladding, and the time spent in the necessary temperature region. An assessment of physical and chemical properties of spent metal fuels and how they interact with proposed storage environments would facilitate a better understanding of the possible degradation mechanisms applicable to these waste forms. This assessment, including an initial characterization of the cladding integrity and of the quantity and thermal characteristics of spent fuel to be loaded in DSSs, would be performed as part of the certification process.

Degradation mechanisms are material-dependent and are functions of environmental, thermal, mechanical, and irradiation conditions such as chemistry, relative humidity, temperature, and radiation level. The current state of materials evaluations for spent metal fuel and TRISO DSSs and DSFs does not completely characterize specific canister and non-LWR fuel performance under all storage conditions and storage environments that could be expected for ARF. Additional characterization of material performance and the influence of the different possible physical and chemical ARF waste forms, possible ARF configurations, radiation effects, thermal loading, and long term canister material performance under a variety of storage environments will clarify the fuel and material performance of ARF and address challenges observed from past storage experience.

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