STORAGE EXPERIENCE WITH SPENT (IRRADIATED) ADVANCED REACTOR FUEL TYPES

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Prepared by

Nathan Hall
Xihua He
Yi-Ming Pan
Patrick LaPlante

Center for Nuclear Waste Regulatory Analyses
San Antonio, Texas

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ABSTRACT

This report presents information associated with storage of spent (irradiated) non-light water reactor (LWR) fuel types based on a review of relevant published operating experience. Non-LWR fuel considered in this review includes solid coated particle fuel, commonly referred to as tristructural isotropic (TRISO), and nuclear metal fuel characteristic of compact fast reactors. For the TRISO spent fuel evaluated, literature documenting the storage experience for Fort St. Vrain (FSV) and Arbeitsgemeinschaft Versuchsreaktor (AVR) was reviewed. For spent nuclear metal fuel, literature documenting the storage experience for the Experimental Breeder Reactor-II (EBR-II), Fast Flux Test Facility (FFTF), and Enrico Fermi Nuclear Generating Station Unit 1 (Fermi 1) was reviewed. The goal of the review was to identify important characteristics of the spent fuels and key factors or mechanisms that may contribute to fuel degradation of spent non-LWR fuel during storage.

Based on decades of coated particle fuel testing simulating in-reactor conditions, a number of potential coated particle failure mechanisms have been identified. These mechanisms include migration of fission product palladium from the fuel kernel during irradiation, which could chemically attack the silicon carbide (SiC) layer of TRISO-coated particles by forming palladium silicides at localized reaction sites and compromise the structural integrity of the SiC layer. These failure mechanisms require high stressors (e.g., temperature, radiation rates, and forces) that are not expected to arise in storage settings of the TRISO-coated particle fuel.

Spent nuclear metal fuel is characterized by its porosity, anisotropic swelling, redistributed constituents, fission products accumulation, and sodium fusion. During reactor operation, pores form in the fuel matrix and the fuel swells under the influence of heat and pressure from accumulation of fission product gases. Heat from the fission reaction melts and fuses sodium into the porous fuel matrix. The temperature gradient drives fuel element redistribution. During irradiation, chemical and mechanical interactions degrade the cladding that houses the fuel. Cladding thinning and cracking can occur at high fuel burnup and high temperatures. These characteristics of spent fuel and cladding are more pronounced for the driver fuels used in compact fast reactors than the blanket fuel because the driver fuel experiences higher irradiation levels. Degradation mechanisms of the nuclear metal fuel during reactor operation bound potential degradation mechanisms during storage.

For storage of these spent fuels, available literature information on operating experience with the spent fuel from the FSV, AVR, EBR-II, FFTF, and Fermi-1 was reviewed to identify storage conditions and fuel performance during storage. There are no records of TRISO-coated particle fuel failure under the storage conditions at FSV. At AVR, the presence of moisture compromised the integrity of the canisters, which in turn caused a release of gaseous radionuclides from AVR spent fuel contained in dry storage canisters within CASTOR casks. Since leak rates from the CASTOR casks were low, release into the environment was concluded to be negligible.

Spent nuclear metal fuel stored in wet and dry conditions in containers experienced degradation because water or moisture was reported to intrude the storage container, due to improper sealing. Degraded cladding permitted water contact with sodium, resulting in reactions producing hydrogen and sodium hydroxide. Some characteristics of spent nuclear metal fuel, in particular the degradation of cladding and the presence and fusion of sodium with other fuel constituents, were observed to challenge performance of storage systems. Due to the extremely reactive nature of sodium, containment integrity is an important factor for storage of

spent metal fuel containing sodium. Some spent metal fuels are being converted to other high-level waste forms without metallic sodium at Idaho National Laboratory. As a result, the approach to storing spent metal fuel will continue to involve storage of both the original spent fuel form and the converted fuel forms.

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FIGURES

TABLES

ABBREVIATIONS/ACRONYMS

ANL	Argonne National Laboratory
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
EBR-II	Experimental Breeder Reactor-II
EFPD	Effective Full Power Days
FCCI	fuel-cladding chemical interaction
FCMI	fuel-cladding mechanical interaction
Fermi 1	Fermi Nuclear Generating Station Unit 1
FIMA	fission per initial metal atom
FPs	fission products
Fs	fissium
FSC	fuel storage container
FSV	Fort St. Vrain
FFTF	Fast Flux Test Facility (FFTF)
HFEF	Hot Fuel Examination Facility
HTGR	high-temperature gas-cooled reactors
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
ISFSI	Independent Spent Fuel Storage Installation
LWR	light water reactor
MFC	Materials and Fuel Complex
MT	metric tons
MVDS	Modular Vault Dry Store
NRC	U.S. Nuclear Regulatory Commission
SiC	silicon carbide
SNF	spent nuclear fuel
SNM	Special Nuclear Material
THTR	thorium high temperature reactor
TRISO	tristructural isotropic

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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.

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1 INTRODUCTION

1.1 Background

As the U.S. Nuclear Regulatory Commission (NRC) staff prepares for regulatory interactions and potential 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* (CFR) 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 associated with the storage, transportation, and disposal of nuclear fuel types that have not been previously considered, such as ARF types. This report examines available published information regarding the characteristics of spent ARF types and associated storage operating experience. A subsequent report will assess possible technical issues that may need to be addressed in safety reviews of storage facilities and cask systems associated with storage of these spent 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 of compact fast reactors.

1.2 Purpose and Scope

This report examines information documenting the experience of spent non-LWR fuel storage that may inform NRC staff in preparing to review future license applications for storage of ARF types. Characteristics of irradiated non-LWR fuel, including irradiation-induced physical and chemical changes, are described, as well as observed and potential degradation mechanisms where applicable. Degradation mechanisms during reactor operation and tests of simulated reactor conditions bound potential degradation mechanisms during storage. Those degradation processes were taken into account while examining the documented domestic and international storage experience. Storage experience applicable to spent fuel integrity during storage, as well as the environment in which the fuel was stored, are documented in this report. A literature search was conducted based on applicable U.S. and international experience with storage for two non-LWR fuel types: solid coated particle and nuclear metal fuel. The objective of the literature review was to identify key factors or mechanisms that may have contributed to degradation of spent non-LWR fuel contained within the storage system. Potential degradation mechanisms are discussed, although they have not necessarily been experienced in the current storage systems. The literature review effort focused on storage experience for Fort St. Vrain (FSV), Arbeitsgemeinschaft Versuchsreaktor (AVR), the Experimental Breeder Reactor-II (EBR-II), Fast Flux Test Facility (FFTF), and Enrico Fermi Nuclear Generating Station Unit 1 (Fermi 1), because information on those systems is publicly available. Information relevant to storage and fuel performance that would be important for future licensing is discussed.

2 CHARACTERISTICS OF SPENT (IRRADIATED) ARF TYPES

2.1 Coated Particle Fuel

The characteristics of unirradiated coated particle fuel were reviewed in the first report in this series (Hall et al., 2019). For spent coated particle fuel, the literature data are somewhat limited except for FSV fuel. The NRC Site-Specific License No. Special Nuclear Material (SNM)-2504 for the FSV Independent Spent Fuel Storage Installation (ISFSI) was issued to store irradiated TRISO-coated particles inside prismatic block fuel elements. Detailed specifications and characteristic of the FSV spent fuel elements are documented in two reports by the U.S. Department of Energy (DOE) (DOE, 2010, 1992) and by Marschman et al. (1993). The following sections provide summaries of the physical and chemical characteristics, thermal characteristics, and radiological characteristics of the FSV spent fuel elements, as well as coated particle fuel degradation mechanisms.

Physical and Chemical Characteristics

There are three types of spent fuel elements stored at the FSV ISFSI: (i) standard fuel elements, (ii) control fuel elements, and (iii) bottom control fuel elements (DOE, 2010, 1992; Marschman et al., 1993). The individual fuel elements are hexagonal graphite blocks drilled with fuel holes and coolant channels. The fuel is in the form of two types of TRISO-coated carbide particles (i.e., fissile and fertile particles[1]) bonded with a carbonaceous matrix into fuel compacts within the fuel holes. Fresh fissile particles contain thorium and uranium enriched to 93.5 percent U-235; fresh fertile particles contain only thorium. The spent fuel elements include mainly thorium, uranium, and mixed fission products, with small amounts of transuranic actinides. An example of the heavy metal content of a fresh and a spent FSV fuel element is listed in Table 2-1 (DOE, 1992). The burnup of the FSV spent fuel element was calculated to be 11.27 percent fissions per initial metal atom (FIMA) for the fissile material and 1.29 percent FIMA for the fertile material, or an average burnup of 32,600 MWd/MT.

Thermal Characteristics

The FSV ISFSI is designed to limit the temperature of the spent fuel elements to less than 399 °C [750 °F] (NRC, 2011; DOE, 2010). This limit is based on the spent fuel element with the highest calculated heat generation rate at 600 days after shutdown. The heat generation rates for a maximum and an average spent fuel element at 600 days after shutdown were calculated to be 150 W and 85 W, respectively. The irradiation period of the fuel element associated with this maximum heat generation rate was assumed to be 945 Effective Full Power Days (EFPD), which is equivalent to a core average burnup of 52,000 MWd/MT (DOE, 2010). These decay heat design values bound all FSV spent fuel elements. The actual average irradiation period of the fuel elements was 230 EFPD, and the actual core average burnup at end-of-life was calculated to be 38,700 MWd/MT. In addition, the in-service time for the ISFSI was 859 days after reactor shutdown. The heat generation rates calculated for a maximum and an average spent fuel element at 859 days after shutdown, using actual burnup, were 101 W and 55 W, respectively (DOE, 2010).

[1]Fertile material is material not directly fissionable by thermal neutrons, but that can become fissionable by neutron absorption and further nuclear conversions.

Table 2-1. Heavy metal content of FSV fresh and spent fuel element (DOE, 1992, Table 4.2.7)				
Radionuclide	Fissile Particle		Fertile Particle	
	Heavy metal content for fresh fuel (g)	Heavy metal content for spent fuel (g)	Heavy metal content for fresh fuel (g)	Heavy metal content for spent fuel (g)
Th-232	1,832.23	1,771.69	8,331.77	8,056.46
Pa-231	0	0.01	0	0.03
U-232	0	0.01	0	0.03
U-233	0	33.60	0	152.78
U-234	3.24	5.30	0	14.19
U-235	407.07	123.40	0	1.58
U-236	1.24	49.72	0	0.10
U-238	25.46	22.58	0	0
Np-237	0	3.44	0	0
Pu-238	0	0.72	0	0
Pu-239	0	0.54	0	0
Pu-240	0	0.24	0	0
Pu-241	0	0.20	0	0
Pu-242	0	0.13	0	0

Radiological Characteristics

The design criteria for acceptable spent fuel radiological characteristics at the FSV ISFSI are the radiation as loaded at 2.97×10^{14} photons/s and 3.31×10^{5} neutrons/s (NRC, 2011; DOE, 2010). The radiological characteristics are based on the analysis with the fuel element irradiated to 52,000 MWd/MT and decayed 600 days. These radiological characteristics bound all FSV spent fuel elements. The radionuclide content of the FSV spent fuel element is not available in any public document. Table 2-2 shows the radionuclide content of a Peach Bottom spent fuel element with a cooling time of 120 days (DOE, 1992). The irradiation period of the fuel element was assumed to be 900 EFPD, which is equivalent to a burnup of 73,000 MWd/MT.

Coated Particle Fuel Degradation Mechanisms

The TRISO-coated particle fuel is the international consensus design for high-temperature gas-cooled reactors based on its superior structural integrity and fission product retention (INL, 2010). A number of coated particle fuel failure mechanisms have been identified under

Table 2-2. Radionuclide content of Peach Bottom spent fuel element (DOE, 1992, Table 4.3.6)

Radionuclide	Activity (Ci)	Radionuclide	Activity (Ci)
Kr-85	50	Ba-137	600
Sr-89	1,170	Ba-140	1.8
Sr-90	393	La-140	2.0
Y-90	393	Ce-141	635
Y-91	1,730	Pr-143	19
Zr-95	2,100	Ce-144	5,060
Nb-95	3,980	Pr-144	5,060
Ru-103	480	Pm-147	1,500
Ru-103m	480	Sm-151	13
Ru-106	400	Pa-233	22,000
Rh-106	400	U-233	42
Te-127m	22	U-234	465
Te-127	22	Pu-238	9.5
Te-129m	31	Pu-239	269
Te-129	31	Pu-240	230
Cs-137	600	Pu-241	20

aggressive in-reactor irradiation and postulated accident conditions as the TRISO-coated particle fuel is exposed to high temperature or attains higher burnup. These failure mechanisms are described in detail in IAEA-TECDOC-1645 (IAEA, 2010), INL/EXT-10-18610 (INL, 2010), and NUREG/CR-6844 (Morris et al., 2004), including

- pressure vessel (i.e., the SiC layer) failure caused by internal gas pressure,
- irradiation-induced cracking and debonding of the pyrocarbon layers,
- fuel kernel migration,
- chemical attack of the silicon carbide (SiC) layer,
- thermal decomposition of the SiC layer, and
- enhanced SiC permeability and/or SiC degradation.

Since the SiC layer is the primary structural component of the TRISO-coated particle fuel, the performance of the SiC layer is critical to the overall fuel performance under in-reactor and accident conditions. For example, migration of fission product palladium from the fuel kernel during irradiation could chemically attack the SiC layer of TRISO-coated particles by forming palladium silicides at localized reaction sites and compromise the structural integrity of the layer. Chemical attack of the SiC layer caused by fission product interactions is evident in Figure 2-1 (INL, 2010).

The above coated particle failure mechanisms identified under in-reactor or postulated accident conditions are not expected to develop during storage, due to lower stressors (e.g., deformation,

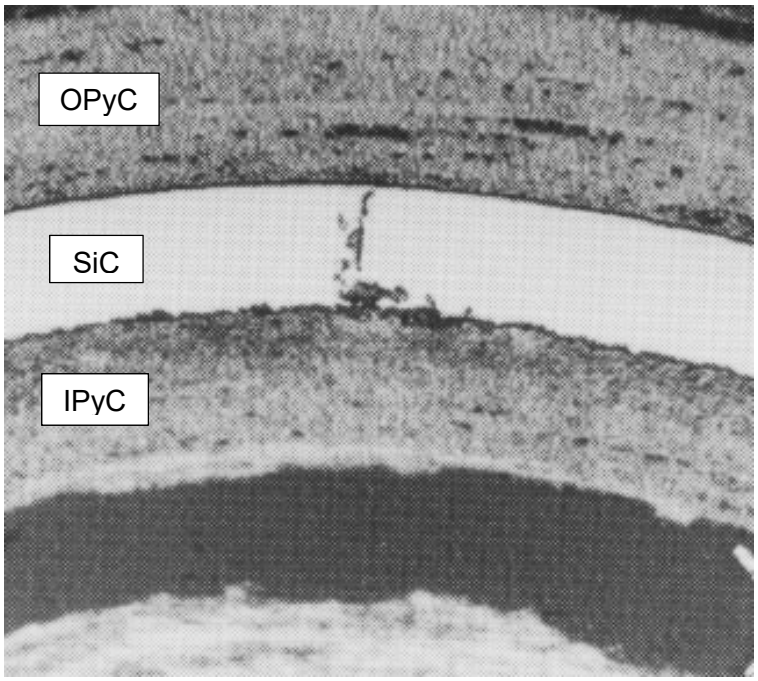

Figure 2-1. Chemical attack of the SiC layer caused by fission product interactions. IPyC is inner pyrocarbon and OPyC is outer pyrocarbon. (INL, 2010, Figure 3)

temperature, radiation rate) in expected storage settings. To date, TRISO-coated particle fuel degradation has not been reported associated with dry storage conditions, as discussed in Section 3 of this report.

2.2 Nuclear Metal Fuel

The characteristics of unirradiated metal fuel were reviewed in the first report in this series (Hall et al., 2019). The following sections provide summaries of the physical, chemical, thermal, and radiological characteristics of the spent metal fuel, particularly the fuel used in FFTF and the driver fuel used in EBR-II and Fermi-1 operation. These fuels all contained more than 65 percent U-235 in the fresh fuel and experienced high levels of irradiation. Irradiation-induced degradation mechanisms are summarized. The blanket fuel used in EBR-II and Fermi-1 operations with depleted uranium (<0.35 percent U-235) in the fresh fuel is not discussed, because the much lower level of irradiation affecting blanket fuel results in insignificant changes compared to the driver fuel.

Physical and Chemical Characteristics

Figure 2-2 shows a schematic diagram of spent metal fuel in comparison to unirradiated (fresh) metal fuel. At burnup lower than approximately 2 atomic percent, fission gases such as xenon and krypton produce micropores inside the irradiated fuel matrix. The fuel swells under the influence of pressure from fission gas and accumulation of solid fission products (FPs). As fuel swells by about 30 percent in volume, the micro-pores begin to connect, forming larger pores and pathways for interstitial fission gas to move upward to the gas plenum. Because of the heat from the fission reaction and the low melting point of sodium, the sodium thermal bond around the fuel in the fuel pin melts and the liquid sodium is displaced upward in the gas plenum by the swelled fuel.

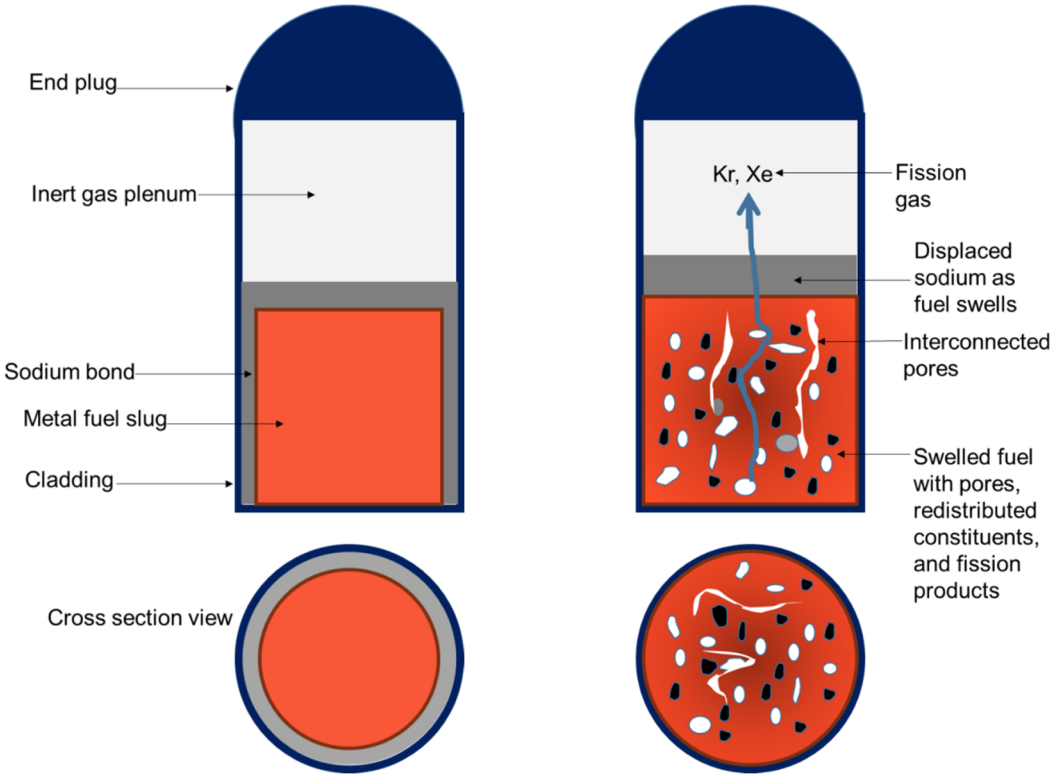

Figure 2-2. Schematic diagram of (a) unirradiated (fresh) metal fuel and (b) irradiated (spent) metal fuel

Some of this liquid sodium enters the interconnected pores within the fuel and fuses with the fuel.

Irradiated fuel was observed to swell in both radial and axial directions, but more in the radial direction due to larger radial temperature gradients (FRWG, 2018). The higher rate of radial swelling creates stress in the fuel large enough to result in the formation of cracks, which could be filled with fuel constituents as irradiation continues. As burnup increases, fuel could swell until it contacts the cladding. The extent of this anisotropic swelling depends on fuel composition. The axial growth of the uranium-based fuels such as U-Fs[2] or U-Zr is in the range of 8-10 percent, whereas it is in the range of 3-4 percent for the ternary fuel (U-Pu-Zr). Because the swelling can compromise the performance of the fuel, designers had been modifying the fuel composition and heat treatment conditions to mitigate the issue. For example, one strategy used to reduce the swelling rate is to add minor amounts of impurities, particularly silicon, to the fuel.

The temperature gradient also drives diffusion of fuel constituents, more in the radial direction than the axial. For U-Pu-Zr fuel, in early stages of irradiation, zirconium tends to migrate to the center and the periphery, uranium tends to migrate in the opposite direction, and plutonium

[2]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.

tends not to migrate. Fuel restructuring can lead to the formation of radial zones such as the example in Figure 2-3. The migration of zirconium can be beneficial to fuel performance

Figure 2-3. U-15Pu-12Zr fuel pin irradiated in EBR-II to 2.4 atomic percent burnup showing interconnected pores and three radial zones due to constituent migration induced by temperature gradients (Kittel et al., 1993) (Pending copyright permission)

because additional zirconium at the center tends to raise the solidus temperature at the peak temperature region and additional zirconium at the periphery tends to improve the fuel-cladding compatibility.

In summary, spent (irradiated) nuclear metal fuels, particularly driver fuel with high enriched U-235, have unique physical and chemical characteristics, specified by porosity, anisotropic swelling, redistributed constituents, FPs accumulation, and sodium fusion. The extent of changes to fuel under the influence of irradiation mainly depends on the composition of the fuel and the burnup. In comparison to driver fuel, the blanket metal fuel (which undergoes much less fission) experiences much less swelling, inter-diffusion between the fuel and cladding, and sodium fusion (Walters et al., 1984).

Thermal Characteristics

During reactor operation, it is observed that the peak fuel temperature is less than 700 °C [1,290 °F] (FRWG, 2018). The temperature of the spent metal fuel out of the reactor is expected to be below this peak temperature, commensurate with decay heat profiles and heat conductivity of surrounding structures distributing the heat. However, specific information on the thermal characteristics of spent nuclear metal fuel was not found in any of the consulted references.

Radiological Characteristics

Detailed information on the radiological characteristics of spent nuclear metal fuel was not found in the consulted references.

Metal Fuel Degradation Mechanisms

During EBR-II reactor operation, fuel-cladding chemical interaction (FCCI) and fuel-cladding mechanical interaction (FCMI) were the two main degradation mechanisms that affected fuel performance and motivated development of new fuel and cladding materials and fuel design improvement. As illustrated in Figure 2-2, during reactor operation metal fuel slug swells and could contact the cladding. FCCI occurs at the fuel–cladding interface where solid state inter-diffusion can occur between fuel, FPs, and cladding constituents, particularly at high fuel burnup and high temperatures between 500 to 700 °C [932 to 1,292 °F]. The available data, mostly on U–Zr and U–Pu–Zr metal fuels, show that FCCI can result in the development of interaction zones, which can contain relatively low-melting phases and brittle compositions formed from elements in fuel (e.g., U, Pu, and Zr), FPs, and cladding (e.g., Fe, Ni, and Cr). The thickness of the interaction zone can be hundreds of micrometers. FCCI can lead to cladding thinning and cracking and penetration of new phases into the fuel. A couple of FCCI examples are shown in Figure 2-4.

Lanthanide FPs, such as La, Ce, Pr, Nd, and Sm, have been observed in relatively high concentrations in the interaction phases because the lanthanide-iron systems have solidus temperatures around 600 °C [1,112 °F], which is lower than the peak temperature the cladding may experience. Elements such as palladium and indium, added to the fuel alloy, had been observed to mitigate FCCI because they combine with these FPs to form phases with higher solidus temperatures. Active research is ongoing to gain further insights into the phases that may develop at the fuel–cladding interface between different metal fuels, such as U–Fs, U–Pu–Zr, U–Pu–Mo, along with different claddings, such as Type 304 and Type 316 stainless steels, through experimental testing, phase diagram construction, and multiscale computer modeling (Keiser, 2012, 2019; Matthews et al., 2017). The research may lead to the development of new cladding materials or barriers on claddings that can be used to mitigate or eliminate the detrimental effect of FCCI on cladding and fuel.

FCMI occurs if the fuel continues to swell after contacting the cladding, which stresses the cladding and can lead to cladding yielding or even breach. During the time during which EBR-II operated, cladding and fuel designs evolved to mitigate the stress experienced by the cladding by increasing the plenum to fuel volume ratio, decreasing the smear density (i.e., the cross-sectional area fraction occupied by the fuel), increasing the thickness of the cladding material, and choosing material with higher strength and better corrosion resistance such as Type 316 SS (Matthews et al., 2017). Compared to FCCI, FCMI is better understood and can be more effectively mitigated. However, as discussed earlier, FCCI can lead to cladding thinning and cracking, which has similar effects on cladding as FCMI.

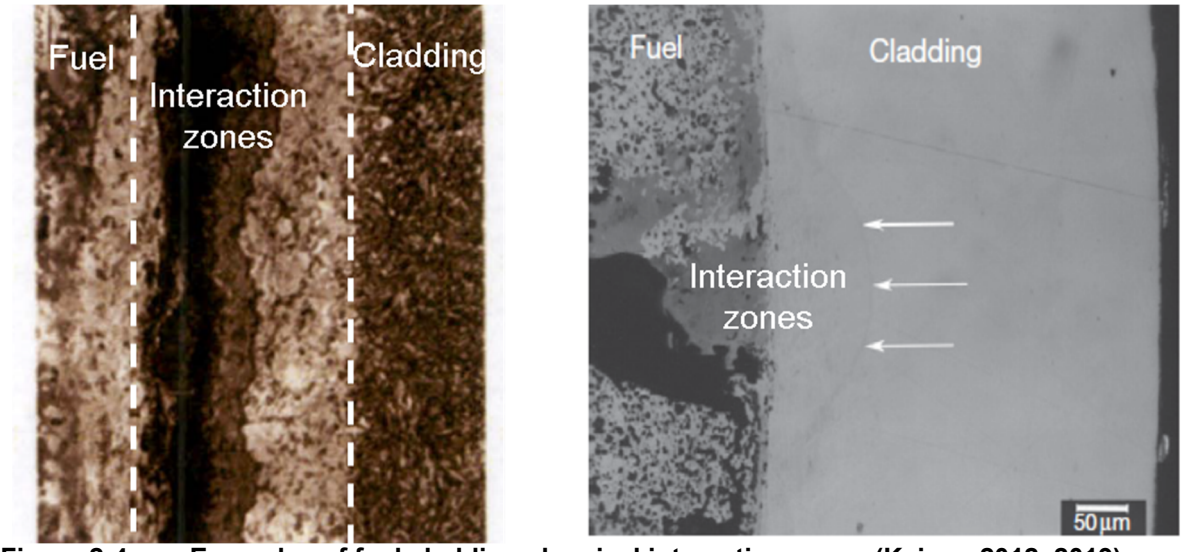

Figure 2-4. Examples of fuel cladding chemical interaction zones (Keiser, 2012, 2019) (Pending copyright permission)

3 STORAGE EXPERIENCE WITH SPENT (IRRADIATED) NON-LWR FUEL

3.1 Coated Particle Fuel Storage Experience

3.1.1 Fort St. Vrain

The FSV reactor was permanently shut down in 1989. A site-specific ISFSI was initially licensed by the NRC in 1991 to store the FSV spent fuel elements. The FSV ISFSI uses the Modular Vault Dry Store (MVDS) system, which is a concrete vault structure that houses spent fuel elements in a matrix of storage positions. Each storage position holds six spent fuel elements in a fuel storage container (FSC). The FSC is a cylindrical carbon steel canister that provides the containment boundary for the stored spent fuel. There are a total of 243 FSCs storing 1,458 fuel elements at the FSV ISFSI (IAEA, 2012; NRC, 2011; DOE, 2010).

The spent fuel elements stored in the FSCs are exposed to an air environment, and decay heat is removed by once-through, buoyancy-driven ambient air flowing across the exterior of the FSCs. The maximum allowable storage temperature of the spent fuel elements is 399 °C [750 °F]. The gamma and neutron sources originating in the fuel are the primary radiation sources. The maximum gamma and neutron radiation levels were calculated to be 2.97×10^{14} photons/s and 3.31×10^{5} neutrons/s, respectively (DOE, 2010).

Under these storage conditions, there are no records of observed aging effects or degradation of FSV spent fuel elements during storage at the FSV ISFSI (NRC, 2011).

3.1.2 Arbeitsgemeinschaft Versuchsreaktor

The AVR operated for 21 years and was shut down in 1988. Nearly 290,000 spherical fuel elements with several types of coated particles were irradiated in AVR. Spent fuel elements for which the desired target burnup (i.e., 95,000 or 150,000 MWd/MT) had been reached were gravity fed from the reactor into stainless steel AVR cans and transferred to wet storage for approximately two years to allow for heat dissipation (IAEA, 1988). From wet storage, fuel pebbles were packed into thin-walled stainless steel dry storage canisters with a capacity of 950 pebbles (Figure 3-1) which occurred within a hot cell facility.

For high burn-up AVR fuel, each dry storage canister has a calculated gamma radiation level of approximately 1.77×10^{3} REM/hr at the canister surface (IAEA, 1980). The dry storage canisters were then packaged using the CASTOR-thorium high temperature reactor (THTR)/AVR casks, made of nodular cast iron. Each CASTOR-THTR/AVR cask weighs about 25 metric tons (MT) and contains two vertically-oriented dry storage canisters, holding 950 pebbles each, or 1900 total pebbles per cask. There are a total of approximately 153 CASTOR-THTR/AVR casks stored at the AVR interim storage facility (Moore, et al., 2014; IAEA, 2012, 2010, 1988; NRC, 2001).

Canisters are sealed with Viton O-ring seals and then pressurized with helium gas (NRC, 2001, IAEA, 1988). Intermediate canister storage occurred in a dry storage facility with heat removal by natural convection. Two canisters were inserted into CASTOR THTR/AVR casks that were closed by a double lid system; the casks are stored in a dry storage facility that is passively cooled. The canister weld and fuel particle coating are credited as barriers against activity release (IAEA, 1988).

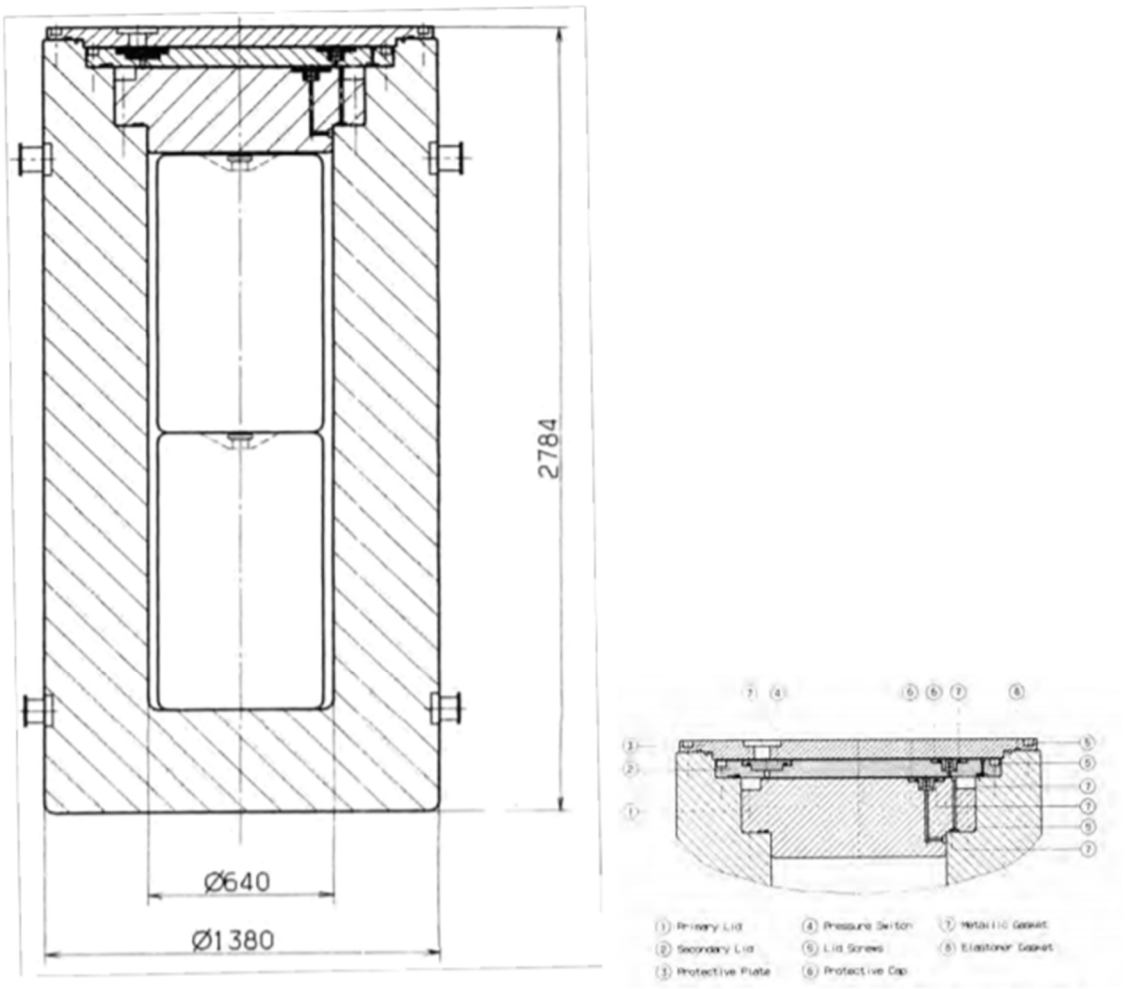

Figure 3-1. Castor-THTR/AVR cask with two dry storage canisters oriented vertically (Moore et al., 2014)

The spent fuel elements stored in the CASTOR-THTR/AVR casks are exposed to an air environment. The maximum allowable fuel storage temperature of the casks for the AVR fuel is less than 70°C [158 °F] (IAEA, 1988; Kirch, 1988). The design heat load for canister/cask systems containing 1,900 pebbles is approximately 7.3 kW (IAEA, 1988).

Steel cans in which AVR spent fuel was initially transferred and stored in the water pool were reported to have leaked. Fuel elements that were found wet were loaded into the same dry storage canister as others and sealed with a leak-tight weld (IAEA 2010). An experimental investigation was conducted during the years 1987–1992 to study possible releases from AVR canisters inside CASTOR casks. During this investigation, radionuclide activity was measured, indicating a minimal release from the canisters to the atmosphere within the CASTOR cask (IAEA 2010). Higher storage temperatures resulted in the release of moisture and tritiated water (IAEA, 2012). Since leak rates from the CASTOR casks were low, release into the environment was concluded to be negligible (IAEA 2010).

3.2 Nuclear Metal Fuel Storage Experience

Operating experience of the five fast reactors [EBR-I, EBR-II, FFTF, Fermi-1, and Dounreay] that used nuclear metal fuel was reviewed in Hall et al. (2019). For the present report, additional literature on spent fuel storage experience at EBR-II, FFTF, and Fermi-1 was found and reviewed. Table 3-1 shows the initial amount of spent nuclear metal fuel in storage in June 1996 (INL, 2007). According to the DOE system for grouping fuels, all of these fuels are in Group 31, which is sodium-bonded spent nuclear fuel (NWTRB, 2017). Some fuel was treated to separate sodium from other components. Table 3-1 also shows the remaining untreated fuel (INL, 2007).

3.2.1 Experimental Breeder Reactor-II

EBR-II had a long operating history, in which a fraction of the spent fuel was reprocessed and refabricated during reaction operation on-site. Other EBR-II spent fuels are in storage at INL.

About 2 MTHM of EBR-II driver spent nuclear fuel (SNF) was originally stored in about 3,600 stainless steel containers in the wet storage basins in one of the five facilities at Idaho Nuclear Technology and Engineering Center (INTEC) at Idaho National Laboratory (INL) since 1978 [Figure 3-2(a)]. Under this wet storage condition, water was found to have leaked into the storage container because of improper sealing and the rupture of degraded cladding, leading to sodium reacting with water producing hydrogen and sodium hydroxide (Pahl, 2000). DOE is in the process of transferring these fuels from wet storage to dry storage at the Radioactive Scrap and Waste Facility, which is one of the two facilities at the Materials and Fuel Complex (MFC) at INL. This facility is a below-grade silo dry storage facility [Figure 3-2(b)]. The fuel storage container has carbon steel as an inner container, stainless steel as an outer container, and another outer layer of carbon steel as a liner. The liners are cathodically protected from corrosion and have shield plugs at the tops to shield radiation and prevent water intrusion. DOE plans to transfer all of the EBR-II driver fuels from wet storage to dry storage by 2022.

Because of sodium infusion into the fuel during operation and the extremely high reactivity of sodium with any moisture, the driver fuel requires chemical treatment before it can be accepted and disposed in any permanent repository. As such, during interim storage, DOE is treating the driver fuel at the Fuel Conditioning Facility at INL using a chemical treatment process developed by DOE (schematically summarized in Figure 3-3). In this process, the spent fuel is chopped into segments before placement into an electrorefiner to allow the molten salt to react with the fuel. The reacted uranium deposits on the cathode from the molten salt in the electrorefiner and is subsequently purified by removing any attached salt in the cathode processor. The purified uranium from the driver fuel with high enrichment is diluted with depleted uranium in a casting furnace to create a uranium product with low enrichment. The cladding and other metals that do not dissolve in the molten salt are removed from the electrorefiner. These removed materials, along with any fission products, are combined with zirconium in a metal waste furnace to generate the metallic waste form. The salt containing transuranic isotopes and fission products is removed from the electrorefiner and blended with zeolite and glass, and the mixture is added to a ceramic waste furnace to produce the ceramic waste form. As such, the chemical treatment process generates a metallic high-level waste form and a ceramic high-level waste form that are stored at the Radioactive Scrap and Waste Facility, awaiting permanent disposal. Table 3-1 shows that as of 2007, only 10 percent of the EBR-II SNF had been treated. The remaining SNF may be treated continuously or placed in interim dry storage.

Table 3-1. DOE sodium-bonded spent nuclear metal fuel in storage (INL, 2007)				
	EBR-II driver fuel (metric tons of heavy metal, MTHM)	**EBR-II blanket fuel (MTHM)**	**FFTF Driver fuel (MTHM)**	**Fermi-1 Blanket fuel (MTHM)**
Initial fuel amount in June 1996	3.1	22.4	0.25	34.0
Remaining untreated fuel reported in 2007	2.3	19.9	0.25	34.0

(a)

(b)

Figure 3-2. EBR-II fuel storage: (a) wet storage basins at Idaho Nuclear Technology and Engineering Center at INL and (b) an aerial view of the Radioactive Scrap and Waste Facility at the Materials and Fuel Complex at INL in the background, and a ground-level photo of the facility with a schematic cutaway of an underground storage vault and container in the foreground (NWTRB, 2017)

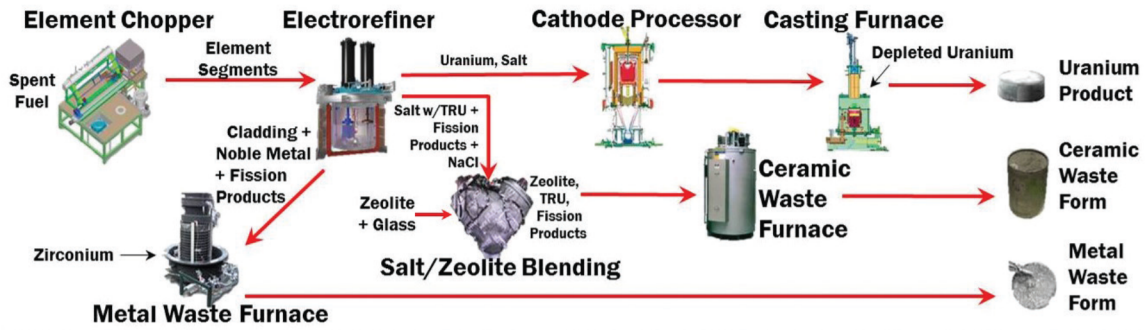

Figure 3-3. Chemical treatment process of sodium-bonded spent nuclear fuel at Idaho National Laboratory (NWTRB, 2017)

The EBR-II blanket fuel is also stored in storage containers at Radioactive Scrap and Waste Facility at MFC [Figure 3-2(b)]. Table 3-1 shows that a small portion of the blanket fuel was treated. DOE is considering physical separation of the sodium or a chemical treatment process to treat the fuel before it can be disposed in a permanent repository.

3.2.2 Fast Flux Test Facility

Wootan et al. (2017) indicate that the unirradiated metal fuel for FFTF was stored in a fuel storage facility within a complex including the reactor and other facilities at the Hanford Site (Figure 3-4). Wootan et al. (2017) further state that failed fuel pins were stored in two fuel storage vessels and cesium leaked out of the vessels. The spent metal fuel from FFTF was initially stored at the Hanford Site. In 2008, all of the 0.25 MTHM spent fuel was transferred to INL. A total of 13.6 kg of spent fuel was held apart for future research in a shielded hot cell in a dry environment at the Hot Fuel Examination Facility (HFEF), which is one of the two facilities at the MFC at INL. All of the remaining 0.25 MTHM inventory was chemically treated at Idaho, following the process in Figure 3-3, and separated into metallic and ceramic forms of high-level radioactive waste.

3.2.3 Fermi-1

Fermi-1 blanket fuel has been stored at INTEC in canisters placed into an underground dry storage system (Figure 3-5) within 14 vertical vaults. The facility consists of two generations of designs with three types of vaults. Each type of vault is built from carbon steel pipes with shield plugs; grouted bottoms were emplaced in wells lined with mild steel. The well of the 2nd-generation design extends above grade to prevent surface water from entering the vault; however, water penetrated the 1st-generation vaults. There is limited publicly available information on the characteristics of the Fermi-1 blanket SNF and on how it is stored in this facility. It is likely that it is stored in the 2nd-generation design because of the enhanced design to prevent water entry. During storage, these vaults are under routine surveillance, hydrogen monitoring, and corrosion monitoring. Because the Fermi-1 spent fuel has unique characteristics, DOE is currently evaluating alternative treatment methods of these stored fuels.

The mentioned storage facilities at INL are not NRC-licensed. As such, definitive information on storage and management facilities, such as safety analysis reports and detailed information about the characteristics of stored fuel, is not publicly available.

In summary, spent nuclear metal fuels from fast reactors have been stored in containers in wet and dry conditions at INL. Under both wet and dry storage conditions, water or moisture

entered either the storage container or the cladding, leading to fuel degradation and sodium reaction with water producing hydrogen and sodium hydroxide. DOE is transferring fuel from wet storage to dry storage. In the meantime, DOE is treating some fuels using a chemical process that generates metallic and ceramic high-level radioactive waste forms that could be disposed in a future permanent geologic repository. The challenges of storing these sodium-bonded spent fuels, as well as their transportation and disposal, will be discussed in upcoming reports.

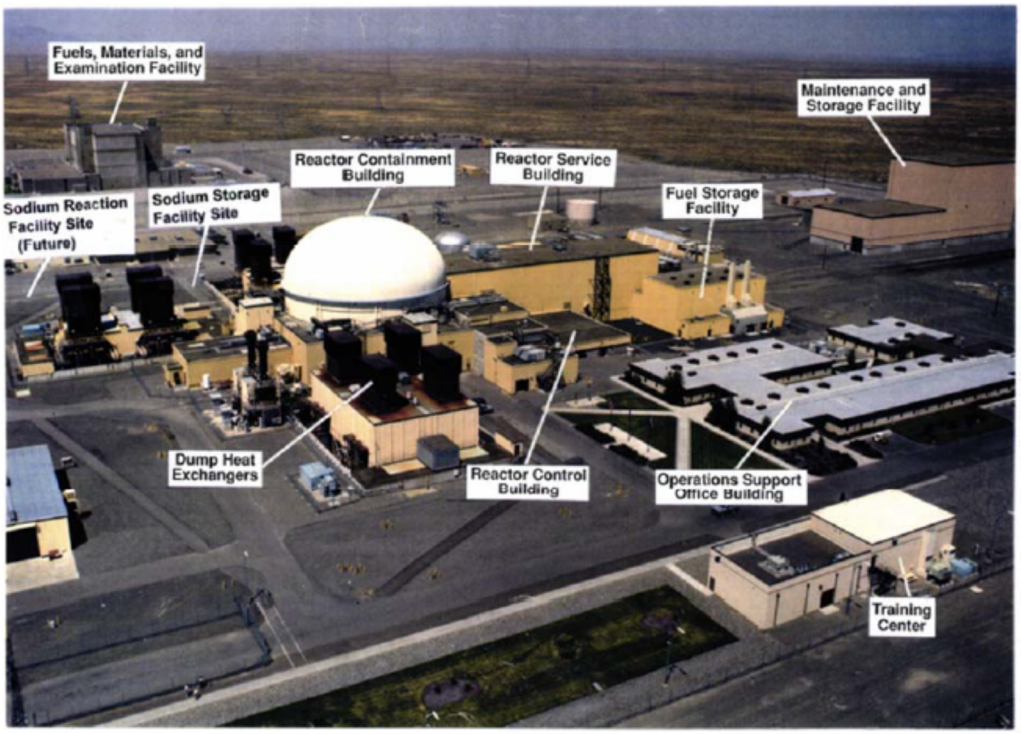

Figure 3-4. Fast Flux Test Facility at the Hanford Site (Wootan et al., 2017). (Pending copyright permission)

(a) (b)

Figure 3-5. Fermi-1 blanket SNF storage at underground vaults at one facility at Idaho Nuclear Technology and Engineering Center at INL: (a) 1st-generation underground vaults built in 1971 and (b) 2nd-generation underground vaults built in 1984 and 1985 (NWTRB, 2017)

4 SUMMARY

This report presented information associated with the storage of spent (irradiated) non- LWR fuel types, based on a review of published relevant experience. Non-LWR fuels considered include solid coated particle fuel, commonly referred to as TRISO, and nuclear metal fuel characteristic of compact fast reactors. The goal of the review was to identify key factors or mechanisms that may contribute to fuel degradation of spent non-LWR fuel during storage. Characteristics of irradiated non-LWR fuel discussed included irradiation-induced physical and chemical changes and thermal and radiological characteristics of the two spent ARF types.

For the non-LWR types evaluated, literature documenting the storage experience for FSV, AVR, and the EBR-II was reviewed. Based on decades of coated particle fuel testing under in-reactor or postulated accident conditions, a number of coated particle failure mechanisms have been identified. These mechanisms include migration of fission product palladium from the fuel kernel during irradiation, which could chemically attack the SiC layer of TRISO-coated particles by forming palladium silicides at localized reaction sites and compromise the structural integrity of the layer. However, these failure mechanisms and processes are not expected to develop during storage of the TRISO-coated particle fuel, because they are associated with higher stresses, temperatures, and radiation fields than in storage settings.

Irradiated nuclear metal fuel is characterized by its porosity, anisotropic swelling, redistributed constituents, FP accumulation, and sodium fusion. During reactor operation, the fuel swells and becomes porous under the influence of heat and pressure from fission gases and other fission products. The thermal gradient redistributes the fuel constituents, and the heat melts the sodium and fuses it in the porous fuel matrix. Cladding that houses the fuel also degrades by fuel-cladding chemical interaction and mechanical interaction. This degradation could lead to cladding thinning and cracking.

For storage of spent non-LWR fuel, available literature information on operating experience with the spent fuel from FSV, AVR, EBR-II, FFTF, and Fermi-1 was reviewed to identify storage conditions and fuel performance during storage. Spent TRISO-coated particle fuel elements are stored at the FSV ISFSI using fuel storage containers housed within the MVDS system and at the AVR interim storage facility using the CASTOR-THTR/AVR casks. There are no records of TRISO-coated particle fuel failure under the storage conditions at FSV. At AVR, the presence of moisture compromised the integrity of the canisters, which in turn caused a release of gaseous radionuclides from AVR spent fuel contained in dry storage canisters within CASTOR casks.

Spent nuclear metal fuel has been stored in wet and dry conditions at INL. The potential interaction of sodium with the storage environment merits special attention. Moisture intruded the storage container because of improper sealing, and degraded cladding permitted water contact with sodium, resulting in production of hydrogen and sodium hydroxide. Some of the stored fuel has been treated chemically to deactivate the sodium, generating metallic and ceramic waste forms suitable for disposal in a permanent repository. The remaining fuel will continue to be treated. As a result, the storage of spent nuclear metal fuel comprises both original waste forms and converted forms. The degradation issues experienced during storage and the changing inventory of waste forms pose potential challenges associated with regulating long-term storage. These challenges will be discussed in the next report.

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