PNNL-35642

Degradation and Failure Phenomena of Advanced Reactor Fuel Concepts

Sodium-Cooled Fast Reactor Metallic Fuel

January 2024

DF Wray
KJ Geelhood

Prepared for the U.S. Nuclear Regulatory Commission
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Abstract

The U.S. Nuclear Regulatory Commission (NRC) is anticipating licensing applications and commercial use of new fuel types in advanced nuclear power reactors that would be designed and built in the United States. Pacific Northwest National Laboratory (PNNL) is providing technical assistance to the NRC related to the newly proposed nuclear fuel and cladding designs that would be deployed in these reactors.

This report focuses specifically on the metallic fuel that is being considered for sodium-cooled fast reactors and specifically on mechanisms that would cause damage or failure to the fuel under reactor operating conditions and design basis accident conditions. There is historic experience with both metallic and oxide fuels in experimental sodium-cooled fast reactors, but this report will focus solely on metallic fuel.

Currently two U.S.-based reactor designers are engaged with the NRC in the application or pre-application review and considering a sodium-cooled fast reactor. TerraPower is engaged with NRC in pre-application review of its Natrium reactor. The current design for this fuel uses a sodium-bonded uranium-10wt% zirconium (U10Zr) fuel clad in HT9 stainless steel. ARC Clean Technology is engaged with NRC in pre-application review of its ARC-100 reactor. The current design for this fuel uses a sodium-bonded uranium-10wt% zirconium (U10Zr) fuel with steel cladding. This report will focus on this fuel system specifically, with broader information given regarding other metallic fuel systems with other stainless steel alloy claddings.

To support the NRC's readiness efforts, this report will identify and discuss degradation and failure modes of metallic fuel concepts for sodium-cooled fast reactors, including fuel performance characteristics that may not be addressed within existing regulatory documents.

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Acronyms and Abbreviations

ANL	Argonne National Laboratory
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ATF	accident tolerant fuel
BCC	body centered cubic
BDBA	beyond design basis accident
BOL	beginning of life
BWR	boiling water reactor
CHF	critical heat flux
CILC	CRUD-induced localized corrosion
CRUD	Chalk River unknown deposit
DBA	design basis accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
EBR-II	Experimental Breeder Reactor II
ECCS	emergency core cooling system
FCCI	fuel cladding chemical interaction
FFTF	Fast Flux Test Facility
GE	General Electric
GEH	General Electric, Hitachi
GNF	Global Nuclear Fuels
HTGR	high temperature gas reactor
INL	Idaho National Laboratory
LOCA	loss-of-coolant accident
LTR	licensing topical report
LWR	light water reactor
MCPR	margin to critical power ratio
MSR	molten salt reactor
NEIMA	Nuclear Energy Innovation and Modernization Act
NRC	U.S. Nuclear Regulatory Commission
OECD-NEA	Organization for Economic Cooperation and Development – Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory
PCMI	pellet-cladding mechanical interaction
PNNL	Pacific Northwest National Laboratory
PRISM	Power Reactor Innovative Small Module

PWR	pressurized water reactor
RIA	reactivity-initiated accident
SAFDL	specified acceptable fuel design limit
SFR	sodium-cooled fast reactor
SRP	Standard Review Plan

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1.0 Introduction

To prepare to review and regulate a new generation of non-light-water reactors (non-LWRs), the NRC staff developed the report "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-light Water Reactor Mission Readiness," issued December 2016.(NRC, 2016) To achieve the goals and objectives stated in that report, the NRC staff developed an implementation action plan dated July 12, 2017.(NRC, 2017) The plan identified specific activities the NRC staff expects to conduct in the near-term (0–5 years), mid-term (5–10 years), and long-term (beyond 10 years) timeframes to achieve non-LWR readiness. The NRC staff has made significant progress on its ongoing activities to fully transition from near-term to mid- and long-term activities to support the licensing of advanced reactors. Many of these activities support readiness requirements in Section 103 of the Nuclear Energy Innovation and Modernization Act (NEIMA), which was signed into law on January 14, 2019 (S.512, 2019)

The United States Nuclear Regulatory Commission (U.S. NRC) is currently engaged with reactor vendors in licensing reviews and pre-application activities for advanced reactor designs. (See https://www.nrc.gov/reactors/new-reactors/advanced/who-were-working-with.html for current status.) The current advanced reactor designs under development fall into one of three categories: molten salt reactors (MSR), high-temperature gas reactors (HTGR), and sodium-cooled fast reactors (SFR). Two of the most developed SFR reactor concepts with metal fuel are the Natrium reactor under development by TerraPower and General Electric, Hitachi (GEH) and the ARC-100 reactor under development by ARC Clean Technology. Additionally, Oklo is working to develop the Powerhouse reactor that relies on fast reactor technology.

As most of the NRC's regulatory framework was developed for the zirconium alloy-clad, UO_2-fueled system, Pacific Northwest National Laboratory (PNNL) is providing technical assistance related to the new proposed fuel and cladding designs that are being developed for advanced reactors to enhance the staff's knowledge base and ultimately support the NRC's efforts to develop and review the required regulatory infrastructure for commercial use of advanced reactors.

The scope of this report includes: A historic perspective on the use of metal fuels in SFRs, an overview of the metallic fuel element that is under consideration for licensing applications in the US (Natrium and ARC-100 reactors) and a discussion of potential degradation and failure modes that should be considered in a safety analysis of this fuel element.

1.1 Background

Cladding for light water reactors (LWRs) has historically been fabricated from zirconium alloys; Zircaloy-2 has been used for boiling water reactors (BWRs) and Zircaloy-4 has been used for pressurized water reactors (PWRs). In-reactor cladding corrosion became an issue as demand for higher burnup levels of LWR fuels grew. To reduce the issue and maintain (or improve) the creep properties of the cladding, nuclear fuel vendors developed proprietary Zr-based alloy claddings that have mostly replaced the traditional Zr-based alloys.

Fuel for LWRs has historically been UO_2 pellets. These pellets are formed by pressing and sintering UO_2 powder to 95-98% of their theoretical density. UO_2 was selected for its excellent radiation stability and its ability to retain a majority of the fission products within the pellets.

However, the thermal conductivity of UO_2 is low which leads to high fuel temperatures and cracking of the pellets.

Metal fuel for SFRs has a similar geometry, but the fuel is typically composed of longer fuel slugs, the fuel cladding gap is filled with liquid sodium, the gap plenum is considerably larger, and the cladding is made of stainless steel. See Figure 1-1.

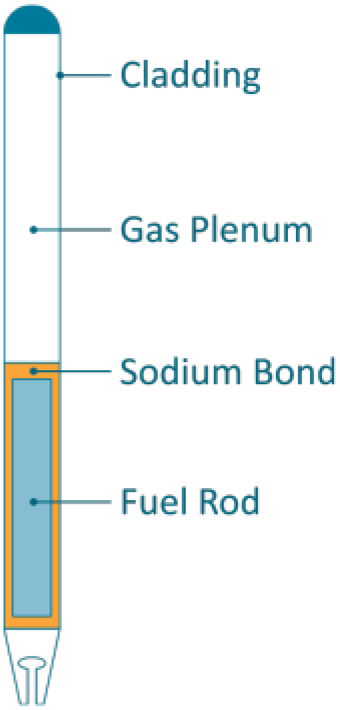

Figure 1-1. Schematic of a metal fuel pin.

The specific damage and failure mechanisms that have historically been identified for LWR fuel will be used as a starting point for SFR metal fuel damage mechanisms and are discussed in greater detail in Section 4.0 of this report. In general for LWR fuel, safety analyses are performed prior to operation to show:

- Rods will not fail (e.g., lose hermeticity, melt fuel pellets, or exceed other design limits) during any condition of normal operation, including during anticipated operational occurrences (AOOs).
- Fuel damage during postulated accidents will not be severe enough to prevent control rod or control blade insertion.
- Fuel failures during postulated accidents will not result in dose outside acceptable limits.
- There is no loss of coolable geometry.

Regulatory Guide 1.232 Appendix B (NRC, 2018) show acceptable design criteria for a sodium cooled fast reactor that are similar for fuel performance to the general design criteria for LWR fuel. It is expected that an applicant will perform similar safety analyses for SFR metal fuel.

1.1.1 Normal Operation and Anticipated Operational Occurrences

Fuel rods in both LWRs and SFRs are static components, yet the cladding is exposed to tensile and compressive stresses and exhibits strain in several directions. For LWR fuel, early in life the fuel/cladding gap is open, and the external pressure is much greater than the internal pressure; the cladding exhibits irradiation-assisted creep in the hoop direction, toward the fuel pellet. At some point, due to the combination of pellet outward swelling and cladding creep, the fuel/cladding gap closes. Continued pellet swelling causes the cladding to strain outward in the hoop direction. Later in life – if enough fission gases are released from the pellet – the internal pressure may exceed the system pressure and irradiation-assisted creep in the hoop direction may cause the fuel/cladding gap to reopen. For SFR metal fuel, the external pressure is often very low (0.1 MPa) so hoop stress will almost always be in the tensile direction. The fuel/cladding gap is filled with sodium, and will eventually close, but because of the sodium bond there will not be a large heat transfer change due to gap closure. The metallic fuel swells considerably more than UO_2 and the swelling is particularly large in the open-gap condition when there is no restraint from the cladding.

For LWR fuel, Zr-based alloy cladding exhibits a hexagonal crystal structure and is highly textured, radiation also causes growth in the axial direction; when the fuel/cladding gap is closed, pellet swelling in the axial direction can result in further cladding strain in the axial direction. HT9 and other stainless steel alloys used in SFR metal fuels have body-centered cubic (BCC) structure so they are not expected to exhibit a preferential crystal orientation that would cause swelling in a particular direction. Additionally, in BCC steel, and particularly BCC steel with no nickel, has been observed to exhibit significantly less radiation swelling than face-centered cubic steel (Garner, Toloczko, and Sencer 2000).

For LWR fuel, the Zr-based alloy cladding reacts with water, and a corrosion layer of ZrO_2 builds up on the cladding's outer surface. In an SFR there is no source of oxygen and the liquid sodium creates a reducing environment. Because of this, there is not expected to be any corrosion products that build up on SFR cladding with irradiation.

LWR and SFR cladding is exposed to the following reactor conditions under normal operations:

Table 1-1. Reactor conditions under normal operations. (US DOE, 2015) (IAEA, 2011) (Flanagan et al, 2014) (Geelhood et al, 2019).

-	BWR	PWR	SFR
Coolant temperature [°F (°C)]	Water: 530-550 (277-288) Steam: 550 (288)	Water: 550-610 (288-321)	Sodium: 932-1022 (500-550)
Coolant pressure [psi (MPa)]	1035 (7.1)	2250 (15.5)	14.4(0.1)
Coolant mass flux [lb/ft^2-hr] For SFR: [gal/min]	~1.05 x 10^6	~2.55 x 10^6	Loop: ~4.35 x 10^4 Pool: ~9 x 10^3
Fast neutron flux [n/m^2-s]	1 x10^{18}	1 x 10^{18}	1 x 10^{19}

Core residence time [days]	1500-2000	1500-2000	2000-2800
Maximum rod-average burnup [GWd/MTU]	62	62	100

The reactor conditions during AOOs are not significantly different than those during normal operation and typically only result in brief changes in power or coolant flow rate. These changes are less than 50% of the nominal values.

1.1.2 Design Basis Accidents

For a LWR, during a DBA event, failure of the cladding is permitted but dose resulting from these failures should not exceed acceptable limits and failure should not impact the coolability of the fuel assembly. This is expected to be the case for SFR based on the similarities in the SFR design criteria in Regulatory Guide 1.232 Appendix B (NRC, 2018) and the general design criteria for LWR's in 10 CFR 100. For LWRs, the main DBAs of interest to the fuel design review are reactivity-initiated accidents (RIAs) and loss-of-coolant accidents (LOCAs), described below. The main DBAs of interest for SFRs have not been formally identified. Design basis event classification for SFRs is an ongoing effort and will be design specific, but there have been previous efforts to do this, including NUREG-1368 (NRC, 1994). In general, historical data has demonstrated that the dominant failure modes for more recent metallic fuel designs in SFRs are cladding breach due to fuel-cladding chemical interaction and fuel melting due to eutectic formation between iron and uranium/plutonium at the fuel-cladding boundary.

1.2 Previous Reviews

No recent publications have been identified that provide a general overview of SFR fuel work. However, two older publications have been identified as providing an overview of early metallic fuel experience in experimental reactors and a set of accident conditions testing that were performed on these fuels:

1. Argonne National Laboratory (ANL) presented a paper at the 1986 International Conference on Reliable Fuels for Liquid Metal Reactors describing early metallic fuel designs and the key fuel behaviors that present a challenge in reaching economical burnups (Porter, 1986).

2. Idaho National Laboratory (INL) published a review paper in *Journal of Nuclear Materials* summarizing the composition, key behaviors, and performance history of experimental metal fuel designs, including failure testing for steady state and transient conditions (Crawford, 2007)

ANL Report

The ANL report gives descriptions of some of the early fuel designs in the EBR-II reactor and describes some of the key fuel behaviors that lead to burnup limitations, as well as documenting some failure mechanisms that are not of particular concern in metal fuel reactors. Some key points from this report include:

- EBR-II initially used U-5Fs as its diver fuel (Fissium is a composite of Mo, Ru, Rh, Pd, and Nb) then switched to a U-10Zr driver fuel during the mid-1980s.

- Advantages of metallic fuels over other fuels include higher thermal conductivity and greater fissile density, as well as ease of fabrication.

- Early fuel designs, such as the Mark-I and Mark-IA, regularly breached around 3% burnup which is attributed to limited gas release at low burnups leading to too much fuel swelling. This problem was addressed in later designs such as the Mark-II by decreasing the "smear density" from 85% to 75%, allowing the fuel to gain enough porosity before fuel-cladding contact to release most fission gas and prevent damaging levels of swelling.

- More advanced designs such as the Mark-II could survive burnups up to 18% and the main source of rupture changed from fuel-cladding mechanical interaction to a buildup of pressure in the plenum due to fission gas, but this type of failure was still unlikely and was addressed by increasing the plenum to fuel volume ratio in the rods.

- Thermal stresses are only significant during transient operations, but overall thermal stresses are similar to other fuel system types.

- Transient conditions can also lead to rapid phase transformation prompting FCMI and create eutectic melting conditions, but in general fuel deformation is sufficient to prevent this. Cladding penetration rates were shown to decrease with burnup due to fission gas and high melting points of intermetallic compounds, which decreases the risk of eutectic melting and shows good fuel reliability. Additionally, more modern designs containing elevated zirconium content may have different FCCI behavior than older designs, such as U-Fs.

- Sodium corrosion effects were somewhat concerning but are mostly a function of cladding and not fuel, so the use of HT9 fuels (as well as D9 for some experiments, even though HT9 was seen as the leading cladding for metal fuels) boosted reliability against corrosion. Additionally, there was no evidence of reduced stress rupture life due to cladding embrittlement, which is attributed to the sodium bond affecting how fission products interact with the cladding.

INL Report

The INL report offers a detailed summary of motivations behind the development and evolution of metallic fuel designs and the testing that has been performed to support the development of metal fuels that can achieve high burnups. This report describes metal fuel compositions and cladding designs for fuels in the Experimental Breeder Reactor I (EBR-I), Enrico-Fermi, the Dounreay Fast Reactor (DFR), and EBR-II, as well as the total experience for each of these designs. The report then goes on to discuss burnup capabilities for early fuel designs and discusses the fuel behaviors that were thought to contribute to the low burnup limits, and these fuel behaviors were selected for further evaluation of limits for design basis accidents (DBAs). Run-Beyond-Cladding-Breach testing was used to determine the limiting factors for fuel rod failure in metal fuels, and to determine the post-failure behavior of the core and rod. This testing determined that modern metal fuel designs have high safety margins during accidents and that when failure occurs in one or few rods the impact on core function is benign. The led to the determination that the most modern metal fuel designs (U-Zr and U-Pu-Zr fuel) had enough experimental data to be used as a driver fuel for future experimental reactors.

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2.0 Past Use of Metal Fuel in Sodium-Cooled Fast Reactors

Metal fuel experiments in SFRs occurred between the 1960's and the mid 1990's. The two reactors that are responsible for most of the metal fuels experimental data are the Experimental Breeder Reactor II (EBR-II) and the Fast Flux Test Facility (FFTF).

2.1 EBR-II

A significant majority of experimental operational data for metal fuels in sodium cooled fast reactors comes from the Experimental Breeder Reactor II. In the EBR-II, the reactor core and primary cooling system are submerged in a sodium pool. EBR-II operated from 1964 to 1994, and research on metal fuels occurred during the last ten years of this period from 1985 to 1994. For this period, data is currently available for 32 metal fuel experiments featuring 61 fuel subassemblies. Peak power for these subassemblies ranged from 32.8 to 63.3 kW/m (Crawford, et al, 2007). Peak burnup for subassembles was as low as 0.48 at.% and as high as 19.8 at.% The peak cladding temperature for these subassemblies ranged from 507°C to 660°C (944.6°F to 1220°F) while the calculated peak fuel temperatures ranged from 639°C to 766°C (1182.2°F to 1410.8°F) (Yacout et al, 2021). Cladding types used for experiments in the EBR-II include 316SS, HT9, and D9. Zirconium weight percentages for fuels were usually about 10% but were as low as 6% and as high as 14%. Plutonium weight percentages ranged from 0% (binary fuels) up to 26%. Coolant flow rate was also measured and varied significantly for different experiments, from as low as ~46 gallons per minute to as high as 119 gallons per minute. Metal fuel experiments in the EBR-II included post-irradiation examination studies into fission gas release and chemistry, irradiated length and weight, contact and laser profilometry, and gamma scans. A large majority of the data used for this document is from EBR-II.

Table 2-1. EBR-II Operation Data. (Crawford et al, 2007) (Yacout et al, 2021)

Years in Operation	1964-1994
Peak Power	32.8-63.3 kW/m
Burnup	0.48%-19.8%
Peak Cladding Temperature	944.6°F to 1220°F
Peak Fuel Temperature	1182.2°F to 1410.8°F
Cladding Types	316SS, HT9, D9
Zirconium Weight Percentage	6%-14%
Plutonium Weight Percentage	0%-26%

2.2 FFTF

The Fast Flux Test Facility was designed and operated by the U.S. Department of Energy and is located at the Hanford site in Washington State. FFTF was a 400-MWt sodium cooled fast reactor and was operated from 1982 to 1992. Significantly less data is readily available regarding irradiation of metallic fuels in the FFTF than in EBR-II since a centralized data source for FFTF has never been created, and data from FFTF makes up only a small fraction of the data used for this document. Data from (Crawford et al, 2007) provides a sample of FFTF operational data. FFTF burnup data is reported in units of GWd/MTHM, ranging from 38 to 143 GWd/MTHM in the sample data. Peak cladding temperatures range from 577°C to 651°C (1070.6°F to 1203.8°F). All reported tests were U-10Zr with HT9 cladding.

Table 2-2. Sample FFTF Operational Data. (Crawford et al, 2007)

Years in Operation	1982-1992
Peak Power	32.8-63.3 kW/m
Burnup	38-143 GWd/MTHM
Peak Cladding Temperature	1070.6°F to 1203.8°F
Cladding Types	HT9
Zirconium Weight Percentage	10%
Plutonium Weight Percentage	0% (Binary Only)

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3.0 Overview of Near-Term SFR Reactors and Fuel

There are currently no SFR submittals under review by NRC. However, NRC has been engaging with TerraPower and GEH on pre-submittal activities for their Natrium reactor. NRC also recently began pre-submittal activities with ARC Clean Technology for their ARC-100 reactor. Details of these reactors and fuel are described in the following sections.

3.1 Natrium Reactor and Fuel

The Natrium reactor is currently being developed jointly by TerraPower and GE. This reactor is similar in design to the GE PRISM reactor that had previously been developed but has higher power than PRISM. The Natrium reactor is a 345MWe sodium-cooled fast reactor combined with a molten salt thermal energy storage system that can boost the system's output to 500MWe of power for more than five and a half hours when needed. The layout of the Natrium reactor is shown in Figure 3-1. Table 3-1 lists parameters for the Natrium reactor.

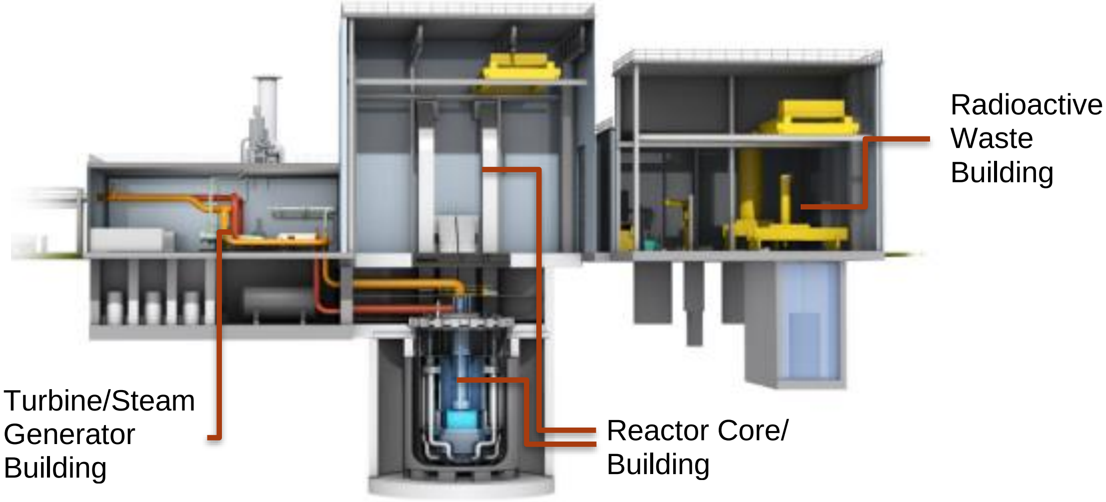

Figure 3-1. Natrium reactor layout. (Natrium Power)

Table 3-1. Natrium reactor parameters (Natrium Power)

Parameter	Value
Type	Sodium cooled fast reactor
Power	820 $MW_{thermal}$ 345 $MW_{electric}$
Coolant Operating Temperature	500°C

The fuel for the Natrium reactor consists of cylindrical uranium metal fuel ingots or slugs that are loaded into long steel tubes sealed on the top and bottom, known as fuel rods. These fuel rods contain a large open space above the fuel slugs to accommodate the gas that is expected to be released from the fuel slugs during the fission process. Additionally, the gap between the fuel slugs and the steel tubes is filled with liquid sodium metal to improve the heat transfer out of the fuel rods.

These fuel rods are grouped into hexagonal arrays and these arrays are placed within an open-ended hexagonal box called a “duct” to form a fuel assembly. Finally, a large number of fuel assemblies is placed in the reactor core and cooled with liquid sodium metal. A diagram of the fuel rods and fuel assemblies are shown in Figure 3-2 (Cheng, 2018). The fuel dimensions and parameters needed for modeling are shown in Table 3-2.

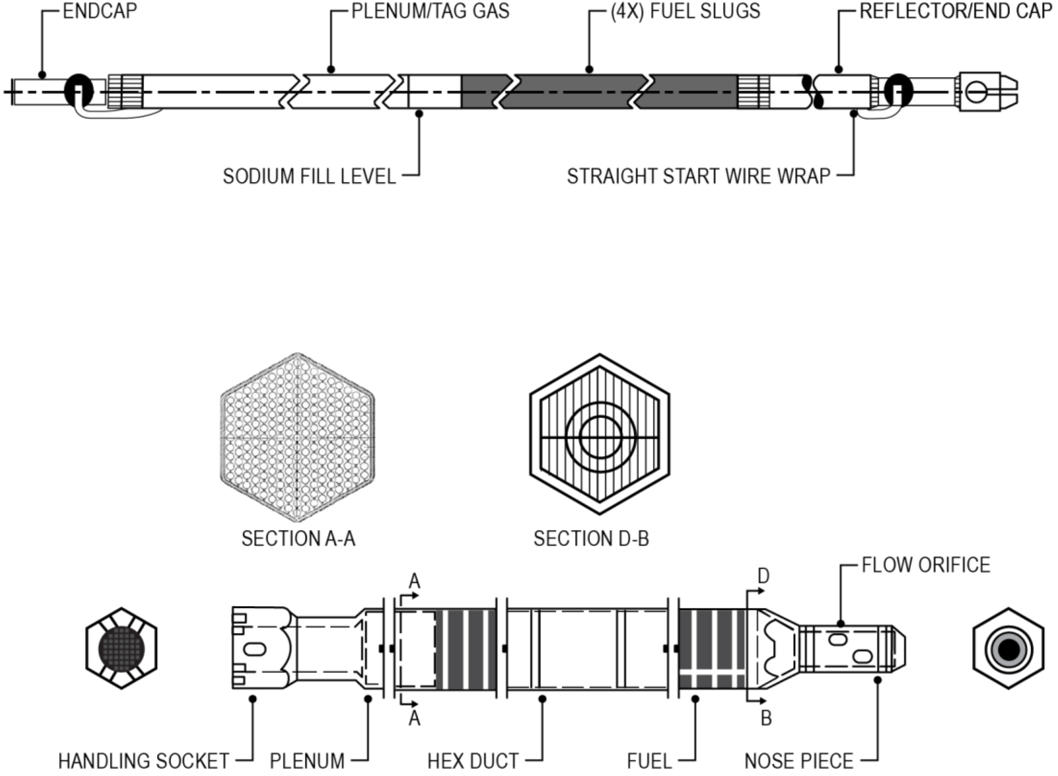

Figure 3-2. Natrium fuel rod and assembly design (Cheng, 2018)

Table 3-2. Natrium fuel parameters. (Cheng, 2018) (Natrium Power)

Parameter	Value
FUEL ROD	
Fuel Type	Metal (U 10 wt% Zr)
Fuel Density	15.5 g/cm^3
Fuel Enrichment (core average)	15.75% ^{235}U235U
Cladding	HT9*
Gap	Liquid sodium
Fuel Slug OD	5.477 mm
Cladding ID	6.322 mm
Cladding OD	7.44 mm
Active fuel length	1016 mm
Total rod length	4070 mm
FUEL ASSEMBLY	
Lattice	Hexagonal
Rods per assembly	271
Assembly length	4724 mm
Rod Pitch (center-to-center distance)	8.92 mm
Duct Material (can surrounding assembly)	HT9
Duct wall thickness	3.94 mm
Duct gap	4.32 mm
REACTOR PARAMETERS	
Assemblies in core	200
Total uranium in core	18 MT
Target burnup	100 GWd/MTU
Average Linear Heat Generation Rate	14.9 kW/m
Burnup in 1 year (with 80% capacity factor)	13 GWd/MTU

* HT9 is a Ferritic/Martensitic steel with nominal composition of Fe (balance), Cr (12 wt%), Mo (1 wt%), W (0.5 wt%), Ni (0.5 wt%), V (0.25 wt%), and C (0.2 wt%)

3.2 ARC-100 Reactor and Fuel

The ARC-100 reactor is currently being developed jointly by ARC Clean Technology. The design foundation of the ARC-100 is rooted in the United States Department of Energy EBR-II program. The ARC-100 reactor is a 100MWe sodium-cooled fast reactor. The layout of the ARC-100 reactor is shown in Figure 3-1. Table 3-1 lists parameters for the ARC-100 reactor.

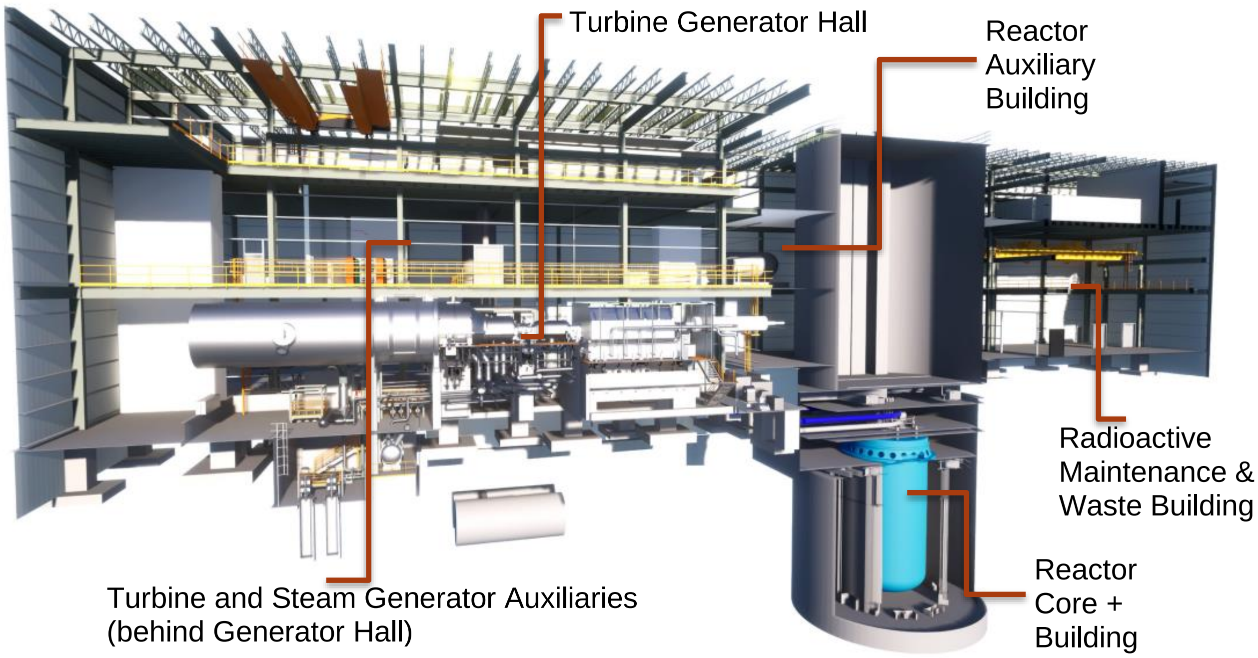

Figure 3-3. ARC-100 reactor layout. (ARC, 2023)

Table 3-3. ARC-100 reactor parameters (ARC, 2023)

Parameter	Value
Type	Sodium cooled fast reactor
Power	286 $MW_{thermal}$ 100 $MW_{electric}$
Coolant Operating Temperature (inlet, outlet)	355°C, 510°C

The fuel for the ARC reactor consists of cylindrical uranium metal fuel ingots or slugs that are loaded into long steel tubes sealed on the top and bottom, known as fuel rods. These fuel rods contain a large open space above the fuel slugs to accommodate the gas that is expected to be released from the fuel slugs during the fission process. Additionally, the gap between the fuel slugs and the steel tubes is filled with liquid sodium metal to improve the heat transfer out of the fuel rods.

These fuel rods are grouped into hexagonal arrays and these arrays are placed within an open-ended hexagonal box called a “duct” to form a fuel assembly. Finally, a large number of fuel assemblies is placed in the reactor core and cooled with liquid sodium metal. A diagram of the fuel rods and fuel assemblies are shown in Figure 3-2. The fuel dimensions and parameters needed for modeling are shown in Table 3-2.

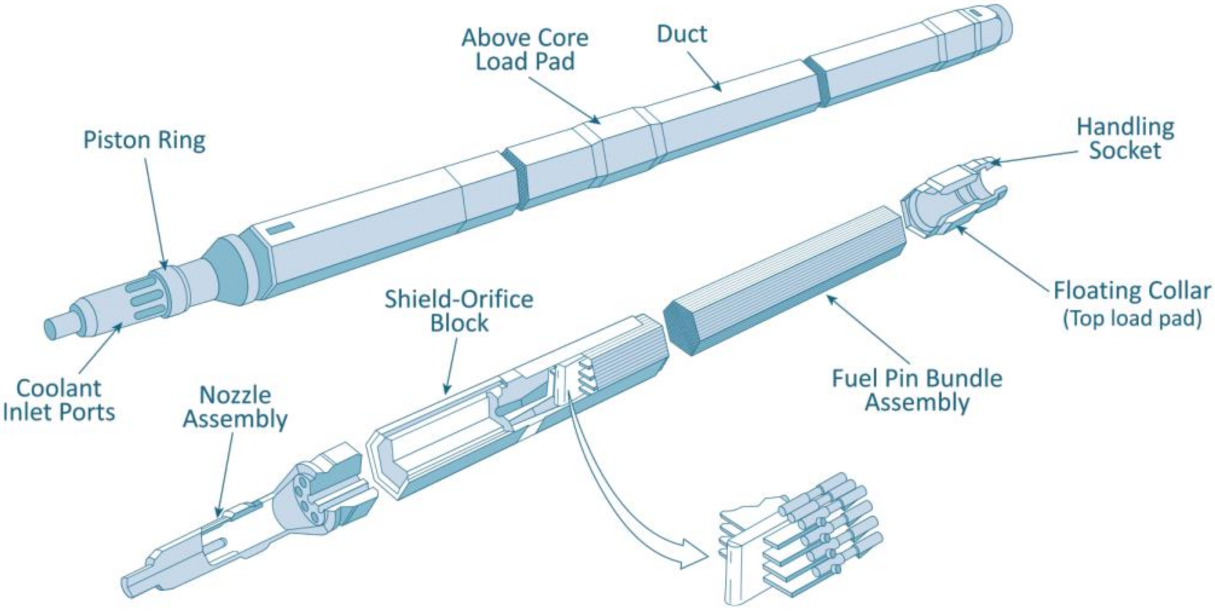

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Figure 3-4. ARC-100 fuel rod and assembly design (ARC, 2023)

Table 3-4. ARC-100 fuel parameters. (ARC, 2010), (ARC, 2023), (Manley et al, 2023)

Parameter	Value
FUEL ROD	
Fuel Type	Metal (U 10 wt% Zr)
Fuel Enrichment (core average)	13.1% ^{235}U
Cladding	HT9
Gap	Liquid sodium
Fuel Slug OD	~1 cm
Cladding OD	12.98 mm
Active fuel length	1500 mm
Total rod length	5860 mm
FUEL ASSEMBLY	
Lattice	Hexagonal
Rods per assembly	217
Assembly length	5860 mm
REACTOR PARAMETERS	
Assemblies in core	99
Total uranium in core	20.7 MT
Target burnup	77 GWd/MTU
Average Linear Heat Generation Rate	13.9 kW/m
Burnup in 1 year (with 80% capacity factor)	4 GWd/MTU
Core Inlet Temperature	355°C
Core Outlet Temperature	510°C

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4.0 Degradation and Failure Mechanisms for Metal Fuel in Sodium-Cooled Fast Reactors

Fuel vendors or licensees typically prepare and submit licensing topical reports (LTRs) to the NRC to describe the codes and methods required to perform bounding or cycle-specific safety analyses for a new fuel assembly design that deviates from limits applied to currently approved methodologies.

This approach to licensing the use of new fuel assemblies can be used as a model for the introduction of a new fuel type into a new reactor system. Although the overall review and approval of the reactor design would be more extensive, for the fuel design, similar to new LWR fuel assemblies, review and approval will be required in the following three areas:

1. Material property correlations to be used in codes for the new cladding.
2. Specified acceptable fuel design limits (SAFDLs) for the new fuel.
3. Methodology and codes used to perform the safety analysis

This section is intended as a guide for the NRC staff as they perform reviews of LTRs related to the use of metal fuel in sodium-cooled fast reactors. This section does not discuss applicable material properties, code or methods, but instead focuses on the failure mechanisms that lead into the development of SAFDLs. Section 4.1 provides an overview of the SAFDLs identified for LWR fuel in Standard Review Plan (SRP) Section 4.2 (U.S. NRC 2007) that should be considered by an applicant. Section 4.2 discusses the applicability of the LWR SAFDLs for metal fuels in sodium cooled fast reactors. Section 4.3 identifies new damage mechanisms that should be considered for metal fuels.

4.1 SAFDL Limits for LWR Fuel

NUREG-0800, Section 4.2, “Fuel System Design” (U.S. NRC 2007) identifies several general phenomena that should be considered for standard LWR fuel and cladding to avoid fuel system damage and fuel rod failure and to ensure fuel coolability. The SRP also provides some general guidance in selecting specific limits in each area. It is the responsibility of the applicant to propose and justify the specific limit to be used in each area and to identify and propose limits for possible damage mechanisms that have not been identified by the SRP.

The SAFDLs mentioned in the SRP are broadly separated into three general categories:

1. Those related to assembly performance, typically addressed by simple calculation, manufacturing controls, and historical data
2. Those related to fuel rod performance, typically addressed for normal operation and AOOs using a thermal-mechanical code
3. Those related to fuel rod performance, typically addressed for accident conditions using a system analysis code with initial conditions provided by a thermal-mechanical code.

Table 4-1 lists each of the SAFDLs mentioned in the SRP of these categories, the purpose of each established limit, and the section of this report the limit is described in more detail.

Table 4-1. SAFDLs from NUREG-0800 and the purpose of each limit (U.S. NRC 2007).

Category	SAFDL	Purpose	Section of this Report the SAFDL is discussed
Assembly Performance	Rod bow	Could impact departure from nucleate boiling ratio (DNBR) or margin to critical power ratio (MCPR)	Section 4.2.1.1
	Irradiation growth	Excessive assembly growth could lead to assembly deformation	Section 4.2.1.2
	Hydraulic lift loads	The weight of the assembly and force of holddown springs should prevent assembly liftoff	Section 4.2.1.3
	Fuel assembly lateral deflection	Lateral deflections should not be so great as to prevent control rods/blades from being inserted	Section 4.2.1.4
	Fretting wear	Excessive fretting wear can lead to failed cladding	Section 4.2.1.5

Category	SAFDL	Purpose	Section of this Report the SAFDL is discussed
Fuel rod performance (normal operation and AOO)	Cladding stress	Prevent failure of cladding from overstress conditions	Section 4.2.2.1
	Cladding strain	Prevent failure of cladding from excessive strain conditions	Section 4.2.2.2
	Cladding fatigue	Prevent failure of cladding from cyclic fatigue	Section 4.2.2.3
	Cladding oxidation, hydriding, and CRUD	Prevent oxide spallation which can result in formation of brittle hydride lens Retain cladding ductility as stated in cladding strain limit	Section 4.2.2.4
	Rod internal pressure	Prevent cladding liftoff due to overpressure during normal operation Prevent reorientation of the hydrides in the radial direction in the cladding which can embrittle the cladding (protect strain limit) Prevent significant deformation resulting in DNB	Section 4.2.2.5
	Internal hydriding	Retain cladding ductility as stated in cladding strain limit	Section 4.2.2.6
	Cladding collapse	Prevent failure of cladding due to collapse in the plenum and at axial pellet gaps which results in large local strains	Section 4.2.2.7
	Overheating of fuel pellets	Prevent centerline melting during normal operation and AOO that would result in large expansion of the fuel pellet that could result in cladding failure.	Section 4.2.2.8
	Pellet-to-Cladding interaction	Prevent failure of cladding from chemically assisted cracking	Section 4.2.2.9
Fuel rod performance	Overheating of the cladding	Failure of cladding and dose consequence if critical heat flux is exceeded	Section 4.2.2.10

Category	SAFDL	Purpose	Section of this Report the SAFDL is discussed
(accident conditions)	Excessive fuel enthalpy	Failure of cladding and dose consequence during RIA if injected energy limit is exceeded; two limits are in place regarding maximum fuel enthalpy to evaluate fuel failure and core cooing	Section 4.2.2.11
	Bursting	Time of burst during LOCA needed to ensure oxidation of inner cladding and associated heat is correctly modeled	Section 4.2.2.12
	Mechanical fracturing	Failure of cladding and dose consequence from external event	Section 4.2.2.13
	Cladding embrittlement	Coolable geometry must be retained following LOCA; there should be no post-LOCA general fuel/assembly failure	Section 4.2.2.14
	Violent expulsion of fuel	Coolable geometry must be retained following RIA; pressure pulse must not damage reactor vessel	Section 4.2.2.15
	Generalized cladding melting	Coolable geometry must be retained following LOCA	Section 4.2.2.16
	Fuel rod ballooning	Degree of ballooning needed to calculate blockage of the coolant channel	Section 4.2.2.12
	Structural deformation	Coolable geometry must be retained following LOCA or seismic event	Section 4.2.2.17

4.2 Applicability of the LWR SAFDLs For Metal Fuels in Sodium Cooled Fast Reactors

This section discusses each of the LWR SAFDLs in the context of metal fuels in sodium-cooled fast reactors. These discussions are at a fairly high level and reflect current understanding which may not consist of sufficient experimental data to justify the use or exclusion of any of these limits. It is the responsibility of the applicant to propose and justify SAFDLs. This information is provided to assist NRC reviewers in the performance of the fuel system safety review.

4.2.1 SAFDLs: Assembly Performance

SAFDLs related to assembly performance are typically performed by simple hand calculations or by citing manufacturing controls or historic data. These limits will need revision relative to those typically used for LWR fuel with Zr-based alloy tubes due to the new materials for fuel and coolant.

4.2.1.1 Rod Bow

Typically for LWR fuel, there is a penalty on departure from nucleate boiling ratio (DNBR) or margin to critical power ratio (MCPR) to account for bowing. It is expected that the limit to sodium coolant boiling will be much greater than DNBR limits in water reactors. However, the rod-to-rod spacing in a SFR fuel assembly is tightly controlled for neutronic purposes and typically wire wraps on the fuel rods are included to ensure proper rod spacing, although tabs on spacer grids have also been considered (Sofu, 2019). Based on this, a SFR fuel assembly may have a bow limit to ensure proper rod spacing for neutronic performance.

4.2.1.2 Irradiation Growth

For LWR fuel, the assembly design allows for a given amount of growth and will define the limit for irradiation growth. Assembly growth limits are typically not as critical in a SFR because there is no top plate to interfere with assembly growth. Nevertheless, a limit for assembly growth should be developed and relevant data should be collected to ensure this limit is not exceeded.

4.2.1.3 Hydraulic Lift Loads

For LWR fuel, the limits for hydraulic lift loads are such that the upward hydraulic forces do not exceed the weight of the assembly and the downward force of the holddown springs. Many SFRs including Natrium are pool-type reactors and although the flow is expected to be lower than under forced convection it should still be determined if there are sufficient hydraulic lifting loads from the flow of molten sodium. Some SFR’s such as ARC-100 are cooled by forced convection of molten sodium, and in this case the lift loads from the sodium circulation could be significant and an assessment should be performed to ensure that these loads do not damage the fuel assemblies.

4.2.1.4 Fuel Assembly Lateral Deflections

For LWR fuel, the limits for fuel assembly lateral deflections are such that control rods (in a PWR) or control blades (in a BWR) can still be inserted as needed. Similar limits should be applied to SFR so that control elements may be inserted as needed. Relevant data should be collected to be used to justify this limit and to ensure this limit is not exceeded.

4.2.1.5 Fretting wear

For LWR fuel, current design limits state that fuel rod failures will not occur due to fretting, which has historically been controlled through debris filters (reducing the possibility of debris fretting) and through spacer design (reducing fretting between fuel rods and grid features). Debris fretting is not likely due to be an issue for SFRs, particularly for pool-type SFRs that have a low coolant flow rate through the reactor. Likewise, grid-to-rod fretting is also not expected to be an issue, again due to low coolant flow rate. For FFTF, designers performed flow tests on simulated fuel assemblies to determine the potential for various wire wraps to cause vibrations. Additionally, FFTF designers allocated 1 mil (25 μm) of the 15 mil (380 μm) cladding thickness to wear, but PIE on the test rod and the reference driver fuel did not show any fretting (Flanagan et al, 2014). However, a surveillance plan is recommended to ensure that unexpected failures due to fuel rod contact with other components or debris are not underestimated.

4.2.2 SAFDLs: Fuel Rod Performance (Normal Operation and AOO)

For LWR fuel, current fuel thermal-mechanical codes informed by material properties can perform the analyses required in this subsection. The geometry for SFR metal fuel is similar to LWR fuel and if appropriate fuel, cladding, and coolant properties are included, an existing code could be applied to SRF metal fuel. However, the limits for the fuel will need revision relative to those typically used for UO_2 fuel with Zr-based alloy tubes. Several of these SAFDLs also have application in accident analysis.

4.2.2.1 Cladding stress

Cladding stress limits are typically set using a method described in Section III of the American Society of Mechanical Engineers (ASME) code (ASME 2017) and are usually based on unirradiated yield stress to represent the lowest yield stress. For stainless steel, including HT9, the use of the unirradiated yield stress should be acceptable to determine a stress limit. Similar to zirconium alloys, irradiation also increases yield stress for steel relative to the unirradiated condition.

4.2.2.2 Cladding strain

For LWR fuel, there are two cladding strain limits that are typically employed. The first is a steady-state limit on the maximum positive and negative deviation from the unirradiated conditions that the cladding may deform throughout life. The second is a transient limit on the maximum strain increment caused by a transient and may also be applicable to accident analysis.

For Zr-based alloys, these cladding strain limits are typically justified based on mechanical tests (axial tension tests and tube burst tests) performed on irradiated cladding tubes. Analysis of FeCrAl alloy cladding have demonstrated that ductility decreases with irradiation in FeCr stainless steel cladding alloys (Field et al. 2017), so these tests are most relevant when performed at the maximum expected fast neutron fluence. The uniform elongation has been typically used as the strain capability for Zr-based alloys (Geelhood, Beyer, and Cunningham 2004). This would be a good metric for stainless steel including HT9 cladding to protect against cladding mechanical failure in SFR.

4.2.2.3 Cladding fatigue

For LWR fuel, the cladding fatigue limit is typically based on the sum of the damage fractions from all the expected strain events being less than 1.0. For Zr-alloy cladding, the damage fractions are usually found relative to the O'Donnell and Langer irradiated fatigue design curve (O'Donnell and Langer 1964). This curve is not applicable to stainless steel tubes. A fatigue design curve should be developed using irradiated samples from the steel alloy to be used in the SFR. Multiple stainless steel alloys may have similar fatigue behavior, but data should be collected from relevant irradiated samples to ensure that this is the case prior to using an existing fatigue design curve.

4.2.2.4 Cladding oxidation, hydriding, and CRUD

The formation of oxide on the surface of SFR fuel cladding during irradiation is unlikely. For LWR fuel, this oxide layer forms from the reaction of water in the coolant with the zirconium. For SFR, the coolant is liquid sodium which is a reducing environment, hence the formation of any oxide on the outer surface of the cladding is not likely.

In LWR fuels, the hydriding failure mode is the failure due to embrittlement of the cladding caused by interaction between the interior cladding wall and a material containing hydrogen (IAEA, 2019). For various reasons, hydriding is not a concern in metal fuels. Firstly, in LWR the source for hydrogen is coolant (water) reacts with the cladding, releasing hydrogen that may be absorbed by the cladding. For metal fuels in a SFR there is no hydrogen in the coolant to cause this problem. Additionally, in zirconium alloys the hydrogen forms a brittle hydride phase while stainless steels typically used in SFR fuel has only a minimal reaction to hydrogen (Cox, 1999). Historically, hydriding also occurred due to a manufacturing error in which small amounts of water got trapped inside the cladding tube, but manufacturing improvements largely eliminated this phenomenon (See Section 4.2.2.6).

With regard to CRUD formation, although it is observed in many LWRs, its definition is "Chalk River Unknown Deposit". It is possible that another unknown deposit could form on fuel rods in a SFR. In FFTF, the 1 mil fretting allowance discussed in Section 4.2.1.5 was also used as a corrosion allowance. To mitigate corrosion in FFTF high purity sodium produced at Savannah River was used in the primary system, although cladding corrosion problems were never observed. Maintaining the purity throughout operation was performed using cold traps to remove impurities and particulates. When tests were run beyond cladding breach, all the fission products were removed, although special cesium traps were installed when the cesium buildup due to running beyond cladding breach became excessive. A surveillance plan is recommended to ensure that a new unknown deposit does not form on SFR cladding. Experience from other SFR's worldwide such as Russian BN-600 and BN-800 could be used to develop plans to mitigate the formation of layers of unknown material.

4.2.2.5 Rod internal pressure

In LWR fuels, rod internal pressure is driven primarily by the reduction in rod void volume due to fuel/clad gap closure and by the release of fission gas (xenon and krypton) from the fuel pellets. In LWR fuel, typically only about 5 to 20% of the produced fission gas is released from the fuel pellets. However, in metal SFR fuel almost all of the produced fission gas is released from the fuel slugs due to the greater diffusivity of these gases in metal than in ceramic. Because of this, SFR metal fuel rods are designed with a large upper plenum volume that can accommodate the complete release of produced fission gas.

Regarding a pressure limit, the external pressure on the rods is very low since typically the coolant is not pressurized, so a conservative limit should be developed to ensure fuel rods are not damaged. Typical LWR limits such as no clad liftoff and no DNB are not appropriate for SFR metal fuel because the fuel rod has been designed with a large plenum to accommodate complete release of fission gas from the fuel and DNB is not applicable in sodium coolant.

Because of the inclusion of a very large plenum and the full release of fission gas, it is likely that a simple hand calculation can be used to demonstrate that the maximum plenum pressure caused by the release of all the fission gas will not result in a rod overpressure situation at end of life when the pressure is the maximum. If an applicant was to take credit for some fission gas retention in the fuel, then a more detailed calculation using a fuel thermal mechanical code with a fission gas release model would be necessary to demonstrate compliance with the limit.

4.2.2.6 Internal Hydriding

In LWR fuels, internal hydriding is typically addressed through manufacturing controls on the pellet moisture limit. Metal fuel slugs are not expected to have any moisture in them and there will likely be surface cleanliness requirements that would limit the water on the surface of the slug. Any water on the fuel surface would react with the liquid sodium in the gap and not be available to the cladding. Additionally, stainless steel has only a minimal reaction to hydrogen (Cox, 1999).

4.2.2.7 Cladding collapse

In LWR fuel, cladding collapse has been mitigated by pellet design features such as dishes and chamfers on the ends of the pellet that effectively eliminate axial gaps in the fuel pellet column.

Cladding collapse is a phenomenon in which densification of the pellets during irradiation causes there to be spacing between pellets (or fuel slugs), and during this the cladding can creep and collapse into the space between the pellets. This is not as much of a concern for metal fuels because 1) the fuel slugs are cast rather than sintered so fuel densification during irradiation is not expected as it is in ceramic pellets which are not fully dense, 2) there are fewer separate fuel segments (slugs are longer than pellets) so there are not as many opportunities for cladding collapse to occur, and 3) the internal pressure is likely always greater than the external pressure since the coolant is not pressurized.

4.2.2.8 Overheating of Fuel Pellets

In LWR fuel, the concern of overheating pellets is due to potential failure due to fuel melting. In metal SFR fuel, the melting temperature is significantly lower than in oxide fuel. However, the thermal conductivity in metal fuel is also much greater than in oxide fuel, so fuel temperatures in metal fuel are lower than in oxide fuel for the same linear heat generation rate. A fuel thermal mechanical code should be used to determine the power to melt for SFR metal fuel.

For SFR metal fuels, in addition to fuel centerline melting from overpower, the fuel and cladding can also potentially fail by forming low temperature eutectics. The concerns about melting from eutectic formation are addressed in Section 4.3.2. For example, a study into the material properties of U-20Pu-10Zr found that the solidus temperature (the lowest temperature at which melting can begin in an alloy) is greater than 1100°C, significantly higher than the discussed temperatures in the eutectic section (Janney et al, 2019).

4.2.2.9 Pellet-to-Cladding Interaction

For LWR fuel, there is typically no explicit limit set on pellet-to-cladding interaction. Various manufacturing designs and inspections and the transient cladding strain limit are expected to cover this SAFDL. For example, early metal fuel pins experienced failures due to high fuel smear densities. In these pins, fuel swelling led to early fuel-cladding contact, which led to excessive cladding stress and strain and eventual failure. Eventually, a smear density around 70% was selected to mitigate these cladding failures through fuel design. For SFR metal fuel, fuel cladding chemical interaction can cause cladding failure and the formation of low temperature eutectics. These are discussed in Sections 4.3.1 and 4.3.2. SAFDLs: Fuel Rod Performance (Accident Conditions)

For LWR fuel, current codes informed by material properties can perform the analyses required in this subsection. The geometry for SFR metal fuel is similar to LWR fuel and if appropriate fuel, cladding, and coolant properties are included, existing codes could be applied to SRF metal fuel. However, the limits for the fuel will need revision relative to those typically used for UO_2 fuel with Zr-based alloy tubes.

NRC regulations and guidance are written specifically for LWR accident analysis. For SFR metal fuel, hypothetical accident conditions and AOO events will have to be developed.

4.2.2.10 Overheating of the cladding

For LWRs, overheating of the cladding refers to exceeding critical heat flux (CHF) and is applicable to AOOs and some accident analyses. Operation above this point results in a reduction of the coolant’s ability to remove heat and can result in cladding damage. For a SFR, the sodium should never boil, so an analysis should be performed to ensure that this does not happen. The boiling point of sodium is 882°C, so as long as the sodium does not boil and the rods remain submerged the cladding will not exceed its melting point of around 1200°C.

4.2.2.11 Excessive fuel enthalpy

For LWR fuel, excessive fuel enthalpy relates to the sudden increase in fuel enthalpy from an RIA below the fuel melting limit which can result in cladding failure due to PCMI. Current fuel enthalpy limits are based on RIA tests that have been performed on irradiated and unirradiated fuel rodlets in various test reactors and a limit of what level of fuel enthalpy increase will cause cladding failure has been determined.

For SFR fuel, some limit should be developed to ensure fuel rods do not failure during a reactivity initiated accident for the types of events that could cause this in a SFR. These limits should be based on test results from relevant fuel segments that show that this energy deposition does not damage fuel.

4.2.2.12 Ballooning and Bursting

For LWR fuel, ballooning and bursting of the fuel rod relates to failure of fuel rods due to high temperatures and high gas pressures during a LOCA (and can also be a consideration during an RIA). It is important to know the rupture stress as a function of temperature and the amount of ballooning that would occur. There are no specific design limits associated with cladding rupture other than that the degree of swelling will not be underestimated and the ballooning will not block the coolant channel Additionally, the time of rupture needs to be known so that oxidation on the cladding inner surface and its associated heat is correctly modeled.

For pool-type SFRs a complete loss of coolant may not be plausible, but accident scenarios should be examined and if there is an event where decay heat cannot be removed and the fuel rod is hypothesized to heat up, then ballooning and bursting should be considered with regard to flow blockage and any oxidation from contact of hot cladding with air if the sodium and cover gas are removed.

If such an event is considered, then relevant high temperature ballooning data should be collected on HT9 or the specific steel alloy to inform models.

4.2.2.13 Mechanical fracturing

For LWRs, mechanical fracturing refers to a defect in the cladding caused by an externally applied force. Typically, this limit has conservatively been set as an applied stressed about 90% of the yield stress. This limit should not be exceeded for normal operation and AOOs. For DBAs, the number of fuel rods exceeding this limit are assumed to have failed and are included in fission product release dose calculations. This limit and a similar analysis are expected to be relevant for SFR metal fuels using the yield stress for the applicable cladding.

4.2.2.14 Cladding Embrittlement

For LWR fuel, cladding embrittlement relates to embrittlement of the fuel cladding, particularly in the ballooned region of the cladding during LOCA. Cladding embrittlement during LOCA should be precluded so the fuel assemblies with ballooned rods are not severely damaged by post-LOCA loads such as reflood and quenching, including blowdown loads. As discussed in Section 4.2.2.12 it is unknown if a SFR would have an accident event similar to LOCA and if it did, it is unknown if there would be any oxidizers present that could react with hot cladding and embrittle it.

For metal fuel in SFR, accident conditions should be critically reviewed to determine if there are any mechanisms that could embrittle cladding during an event with a loss of coolant.

4.2.2.15 Violent Expulsion of Fuel

For LWRs, violent expulsion of fuel relates to the sudden increase in fuel enthalpy from an RIA that can result in melting, fragmentation, and dispersal of fuel. This could result in a loss of coolable geometry and produce a pressure pulse that could damage the reactor vessel.

For SFR metal fuel this phenomenon has not been greatly studied, though the fuel expulsion itself could be less likely than in LWRs because the fuel slugs are much larger than LWR pellets and the ductile metal slugs are less likely to fragment into many small pieces than brittle UO_2 ceramic pellets. However, there is likely some increase in enthalpy that will cause the metallic fuel to melt which would result in a violent expulsion of fuel. Experimental tests should be performed to determine a limit for violent expulsion of fuel in SFR metal fuel during an RIA.

4.2.2.16 Generalized Cladding Melting

For LWRs, generalized cladding melting is applicable to DBAs and is set to preclude the loss of coolable geometry. The limit is set as the cladding melting temperature, which for Zr is 1852 °C. For SFR metal fuel, the limit should be set at the specific melting point for the cladding alloy. For

LWRs there are many limits that are more limiting than the cladding melting limit and this is likely to be the case with SFR metal fuel as well, given limits placed on sodium coolant boiling.

4.2.2.17 Structural deformation

Structural deformation refers to externally applied loads during a LOCA or safe shutdown earthquake that could deform the fuel assemblies or cause fuel fragmentation such that coolable geometry would be lost. This limit has conservatively been set as applied stresses above 90% of the irradiated yield stress. For DBAs, the number of fuel rods exceeding this limit are assumed to have failed and are included in fission product release dose calculations. This limit is acceptable for SFR metal fuel cladding given that the irradiated yield stress is obtained for the cladding.

4.3 New Damage Mechanisms

This section identifies additional damage mechanisms that should be considered for metal fuels in sodium-cooled fast reactors, which may either be addressed by applicants through existing limits or as separate limits. The following damage mechanisms have been identified through a technical review of recent data and a general understanding of fuel and cladding behavior. Each is a physical mechanism that should be addressed even if no credit for performance is credited in the fuel system safety review.

4.3.1 Fuel Cladding Chemical Interaction (FCCI)

There are a few components to the FCCI behavior in metallic fuels. Essentially, FCCI results in the transport of various elements at the fuel-cladding interactive region. Specifically, there is a concern over the transport of fuel elements and fission products into the cladding and transport of the cladding elements into the fuel.

Some of the most in-depth post irradiation tests of FCCI in metallic fuels were from experiments in the Experimental Breeder Reactor II (EBR-II). The data that will be analyzed below is retrieved primarily from an Argonne National Laboratory (ANL) report (Keiser, 2006). This report analyzes a set of experiments in which twenty different fuel elements were irradiated in EBR-II, most of which were either U-23Zr or U-16Pu-23Zr, where the numbers preceding an element specify the atom percentage or at%. Besides fuel composition, these fuel elements varied by cladding material, burnup, and fuel-clad interface temperature at beginning of life (BOL). These rods are described in Table 4-2. From these experiments, there are two significant outcomes for each fuel element with regards to likelihood of failure

Table 4-2. EBR-II rods subjected to PIE.

Fuel ID	Cladding Material	Fuel composition (at%)	Burnup (at%)	Temp at Fuel/Clad Interface at BOL (°C)	Maximum Zone Thickness (μm)	Fuel Failure
DP81	HT9	U-23Zr	5.0	660	70	-
DP11	HT9	U-23Zr	10.0	660	90	-
DP04	HT9	U-23Zr	10.0	660	90	-
DP70	HT9	U-23Zr	10.0	660	140	Yes
DP75	HT9	U-23Zr	10.0	660	170	Yes
T459	HT9	U-16Pu-23Zr	3.0	-	10	-
DP16(1)	HT9	U-16Pu-23Zr	9.7	540	40	-
DP16(2)	HT9	U-16Pu-23Zr	10.1	550	40	-
DP21	HT9	U-16Pu-23Zr	11.4	-	-	-
C709	D9	U-23Zr	9.3	650	111	-
T225	D9	U-23Zr	10.0	-	25	-
T141	D9	U-23Zr	11.9	-	17	-
T159	D9	U-16Pu-23Zr	3.0	-	-	-
XX	D9	U-16Pu-23Zr	6.0	-	75	-
T042	D9	U-7Pu-23Zr	6.0	540	20	-

T087	D9	U-16Pu-23Zr	10.0	-	50	-
A850	D9	U-16Pu-23Zr	10.1	550	100	-
T112	D9	U-16Pu-23Zr	11.9	-	72	-
T106	D9	U-16Pu-23Zr	17.0	-	20	
T341	316SS	U-16Pu-23Zr	0.4	-	-	

Cladding Breach: Only two of the twenty fuel elements experienced a cladding breach, these elements were labelled DP70 and DP75. They were exactly alike in terms of the controlled parameters (Fuel composition is U-23Zr, Cladding Material is HT9, Burnup at.% is 10.0, and interface temperature at BOL is 660 °C). However, two more fuel elements that did not breach, DP11 and DP04, also had the exact same controlled parameters. Another fuel element, DP81, had the same controlled parameters except for burnup, which was lower at 5.0. DP81 did not have a cladding breach. Three more fuel elements (C709, T225, and T141) had the same fuel composition, similar burnup (9.3-11.9 at%) and similar BOL temp (650 for one, but not determined for some others) but had a different cladding material (D9) and these elements did not have a cladding breach. No direct comparison can be done for breach of elements containing Pu, since none of these elements breached and any comparisons to the U-23Zr elements that did not breach are complex because there are at least two parameter differences for any possible combination. A study into the fuel performance of the Mark II fuel for the EBR-II, which consisted of U-Fs fuel instead of U-Zr, found that fuel could reliably be irradiated to at least 10 at.% burnup without experiencing cladding breach, and that some rods reached as high as 16 at.% before breaching (Walters and Kittel, 1980). A later study of “off-normal” transient behavior in metallic fuel reactors found that metallic fuels irradiated in EBR-II did not experience cladding breach at lower burnups when undergoing increased temperature transients (Seidel, et al, 1986). By 1986, some experiments in the EBR-II found that metal fuels could survive up to 18.5 at.% burnup before reaching cladding breach due to FCMI, but most did breach at lower burnups (Porter et al, 1986). With the most recent metal fuel designs, fuel may likely attain a burnup of about 10 at.% without risk of failure due to fuel cladding interaction, but burnups higher than that could pose higher risk

Maximum Interaction Zone Thickness: The interaction zone thickness is the width of the area where there is penetration into the fuel and cladding by elements of the other material. This is an important analysis point because, where there are only a few examples of elements that have failed during irradiation in the EBR-II, the interaction zone thickness is considered a possible indicator of failure likelihood. One analysis in support of this is a comparison of the similar fuel elements that breached in the experiment. The two fuel elements that breached had a much larger zone thickness (140 and 170 microns) than the two fuel elements with the same controlled parameters that did not breach (90 microns). However, it is notable that another analysis of this data (Pahl, et al, 1993) claims that the increased isotope penetration could be due to the breach, instead of the breach being caused by enhanced isotope penetration. Unfortunately, some of the relevant data from these experiments was not recorded: the BOL temperature data was not recorded for most fuel elements with D9 cladding, and the maximum zone thickness was not recorded for a few D9 clad elements as well. Notably, the fuel rods in this analysis were studied across multiple experiments, but this could also be due to the lack of interest in long-term usage of D9 cladding for sodium cooled metal fuel reactors. A paper that analyzed interest in prospective fuels for sodium cooled reactors (Crawford, et al, 2007) remarks on the usage of D9 clad irradiation in the EBR-II: “Although the program included fuel testing in the D9 alloy, deployment of HT9 as the cladding and duct material was the long-term intention.” Some zone thickness comparisons can still be made, however, including some outcomes that would support the use of HT9 cladding over D9 cladding despite the lone two failures occurring in HT9 clad rods. Fuel element C709 had similar BOL temp and burnup to two of the non-breached elements DP11 and

DP04 (9.3at% vs 10.0at% and 650°C vs 660°C) but element C709 had D9 cladding instead of HT9. The C709 had a max zone thickness of 111 microns, as opposed to 90 microns for the other two fuel elements. A similar analysis can be made for a pair of U-16Pu-23Zr fuel elements. Fuel elements DP16(1), DP16(2), and A850 all have similar burnups (9.7, 10.1, and 10.1 respectively) and BOL temps (540, 550, and 550°C) but A850 has D9 clad while the other two have HT9 clad. The A850 also had a much larger zone thickness (100 microns) than the other two elements (40 microns). These two analyses show a larger zone thickness resulting from D9 cladding vs HT9 cladding in the absence of a breach, which would indicate that HT9 is a preferable cladding to prevent failure due to fuel cladding interaction if there is a relationship between zone thickness and probability of failure. The biggest chemical difference between these two claddings is that D9 contains a significant amount of nickel and more chromium than HT9. Iron has been considered the main cladding element penetrating the fuel during FCCI, but nickel has been found to reach significant penetration as well (Harp, et al, 2020). It should be noted that the largest zone thickness recorded for a non-breached fuel element was 111 microns, whereas both breached elements had a thickness of at least 140, though again the cause and effect are not known for certain.

4.3.2 Eutectic Formation

The important aspect of eutectic formation with regards to metal fuels is a low eutectic melting point, where the combination of isotopes from the fuel (U-Zr or U-Pu-Zr) and the cladding (stainless steel, generally consisting of primary components Fe, Ni, Cr) form a material that has a lower melting point than either of the separate materials. This can be a result of isotope transport during FCCI, and if melting occurs then this can exacerbate FCCI even further as the liquid state promotes easier isotope transport. One analysis shows that a range for a temperature threshold can be established for eutectic formation(Cohen, et al, 1993). This study found that, for this fuel composition and cladding combination, the threshold for melting at 5.6% burnup is 740-770°C and for 11% is 650-675°C. HT9 is considered the superior cladding for preventing FCCI, but we do need to consider the effects of varying Plutonium content for this metric. The Mark II fuel design study (Walters and Kittel, 1979) found that sodium cooled fast reactors are very unlikely to reach eutectic temperatures during normal operation, but that eutectic temperatures could be encountered during high temperature transients. However, at temperatures near the eutectic, the fuel-cladding penetration is very small, and it is only temperatures well above the eutectic temperature that result in significant penetration. This study found the eutectic temperature of U-15Pu-10Zr to be around 850°C.

4.3.3 Defects

It is currently unknown what sort of manufacturing checks will result in quality tubes or fuel slugs for this fuel type. As such, it is unknown what defects, let alone their size and concentration, will be introduced in the manufacturing process. Each process in question should define the allowable defects and justify their presence based on testing of cladding with similar defect concentrations.

There is ongoing work by the Department of Energy to develop capabilities to fabricate metallic fuels and determine how to qualify new fuel systems (INL: Materials and Fuels Complex).

5.0 Conclusions

The U.S. Nuclear Regulatory Commission is currently engaging in pre-application reviews of sodium-cooled fast reactors that use metal fuel (Natrium and ARC-100 reactors)

Damage mechanisms for SFR metal fuel that will result in degradation or failure for the fuel have been determined based on the specified acceptable design limits for LWR fuels as well as a literature review of other damage mechanisms that have been observed in this class of fuel.

New damage mechanisms were identified in Section 4.3 and include:

1. Fuel Cladding Chemical Interaction. If burnup is limited to 10 at%, then failure due to this mechanism is unlikely. Additionally, failures are not observed when the interaction zone is less than 110 μm.
2. Eutectic Formation. A temperature threshold should be established that is dependent on fuel composition and fuel burnup.
3. Defects, introduced in the manufacturing process, should be justified based on testing of cladding with similar defect concentrations.

Experience with SFR metal fuel is extremely limited to operation in two experimental reactors (EBR-II and FFTF). Although considerable post-irradiation examination was done on rods irradiated in this reactor, there is no in-situ data such as fuel temperature or gas pressure that can be used to qualify code predictions. Additionally, there is limited data regarding off-normal and accident conditions. An applicant should present a robust set of qualification data for normal, off-normal, and accident conditions based on the specific fuel and cladding that will be used.

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6.0 References

ARC Clean Technology. 2010. "A Sustainable, Cost-Effective Energy Solution for the 21st Century." Advanced Reactor Concepts, LLC.

ARC Clean Technology. 2023. "ARC-100 Technical Summary." Advanced Reactor Concepts, LLC.

ASME. 2017. “Boiler and Pressure Vessel Code.”.

Cheng, L. 2018 "Phenomena Important in Liquid Metal Reactor Simulations," Brookhaven National Laboratory BNL-207816-2018-INRE

Cohen, A.B., Tsai, H., Neimark, L.A. 1993. "Fuel/Cladding Compatibility in U-19Pu-10Zr/HT9-Clad Fuel at Elevated Temperatures." *Journal of Nuclear Materials*, 204: 244-251.

B. Cox. 1999. "Corrosion and Hydriding of Zircaloy Fuel Cladding, The Inside (and Outside) Story." Centre for Nuclear Engineering, University of Toronto.

Crawford, D.C., Porter, D.L., Hayes, S.L. 2007. "Fuels for Sodium-Cooled Fast Reactors: US Perspective." *Journal of Nuclear Materials* 371: 202-231.

EPRI. 2019. “Accident-Tolerant Fuel Valuation: Safety and Economic Benefits (Revision 1).” 3002015091.

Field, Kevin G., Samuel A. Briggs, Kumar Sridharan, Richard H. Howard, and Yukinori Yamamoto. 2017. “Mechanical Properties of Neutron-Irradiated Model and Commercial FeCrAl Alloys.” *Journal of Nuclear Materials* 489: 118–28.

Flanagan, G.F., Mays, G.T., Madni, I.K. 2014. "NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors." NRC and Oak Ridge National Laboratory.

Garner, F. A., M. B. Toloczko, and B. H. Sencer. 2000. “Comparison of Swelling and Irradiation Creep Behavior of Fcc-Austenitic and Bcc-Ferritic/martensitic Alloys at High Neutron Exposure.” *Journal of Nuclear Materials* 276: 123–42.

Geelhood, K.J., C. Beyer, and M. Cunningham. 2004. “Modifications to FRAPTRAN to Predict Fuel Rod Failures Due Tto PCMI During RIA-Type Accidents.” In *2004 International Meeting on LWR Fuel Performance*.

Geelhood, K.J., Luscher, W.G. 2019. "Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts." Pacific Northwest National Laboratory

Harp, J.M., Capriotti, L., Porter, D.L., Cole, J.I. 2020. "U-10Zr and U-5Fs: Fuel/Cladding Chemical Interaction Behavior Differences." *Journal of Nuclear Materials* vol. 528.

IAEA. 2011. "Status and Trends of Nuclear Fuels Technology for Sodium Cooled Fast Reactors." *IAEA Nuclear Energy Series No. NF-T-4.1*

IAEA. 2019. "Review of Fuel Failures in Water Cooled Reactors (2006-2015)."

INL. "Materials and Fuels Complex: Metallic Fuel Line." https://mfc.inl.gov/SitePages/Instruments/Fuels%20and%20Applied%20Science%20Building/Metallic%20fuel%20line.aspx

Janney, D.E., Hayes, S.L., Adkins, C.A. 2019. "A Critical Review of the Experimentally Known Properties of U-Pu-Zr Alloys. Part 1: Phases and Phase Diagrams." Idaho National Laboratory.

Keiser, D.D. 2006. "Fuel-Cladding Interaction Layers in Irradiated U-Zr and U-Pu-Zr Fuel Elements." Argonne National Laboratory ANL-NT-240.

Manley, M., Thompson, P., Pilkington, B. 2023. "Fuel Cycle for ARC-100 Commercial Demonstration at the Poitn LePreau Nuclear Site in New Brunswick." ARC Clean Technology, New Brunswick Power.

Natrium Power. "Reactor Technology." https://natriumpower.com/reactor-technology.

NRC. 1994. "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor."

NRC. 2016. "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission readiness."

NRC. 2017. "NRC Non-Light Water Reactor Near-Term Implementation Action Plans."

NRC. 2018. "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors."

O'Donnell, W., and B. Langer. 1964. “Fatigue Design Basis for Zircaloy Components.” *Nuclear Science and Engineering*, 1–12.

OECD-NEA. 2018. “State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels.” NEA No. 7317.

Pahl, R.G., Lahm, C.E., Hayes, S.L. 1993. "Performance of HT9 Clad Metallic Fuel at High Temperature." *Journal of Nuclear Materials*.

Porter, Hofman, Seidel, Walters. 1986. "Factors Controlling Metal Fuel Lifetime." *International Conference on Feliable Fuels for Liquid Metal Reactors*

S.512 - 115th Congress (2017-2018): Nuclear Energy Innovation and Modernization Act. (2019, January 14). https://www.congress.gov/bill/115th-congress/senate-bill/512

Seidel, B.R., Batte, G.L., Lahm, C.E., Fryer, R.M., Koenig, J.F. 1986. "Off-Normal Performance in EBR-II Driver Fuel." *International Conference on Reliable Fuels for Liquid Metal Reactors*.

Sofu, Tanju. 2019. "Fast Reactor Fuels." *Fast Reactor Technology Training, U.S. Nuclear Regulatory Commission.*

U.S. DOE. 2015. "Sodium-cooled Fast Reactor (SFR) Technology and Safety Overview." https://www.nrc.gov/docs/ML1504/ML15043A307.pdf

U.S. NRC. 2007. “Standard Review Plan: Section 4.2, Fuel System Design.” NUREG-0800, Rev. 3.

Walters, L.C., Kittel, J.H. 1979. "Development and Performance of Metal Fuel Elements for Fast Breeder Reactors." *Nuclear Technology*, 48:3, 273-280.

Yacout, A.M., Mo, K., Oaks, A., Miao, Y., Sofu, T., and Mohamed, W. “FIPD: The SFR Metallic Fuels Irradiation & Physics Database.” *Nuclear Engineering and Design,* vol. 380, p. 111225, 2021. doi: 10.1016/j.nucengdes.2021.111225

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