

Fuel Performance Considerations and Data Needs for Burnup above 62 GWd/MTU

**In-Reactor Performance, Storage, and
Transportation of Spent Nuclear Fuel**

November 2019

KJ Geelhood

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Pacific Northwest National Laboratory
Richland, Washington 99352

Abstract

The U.S. Nuclear Regulatory Commission (NRC) is preparing for anticipated licensing applications requesting burnup extension for light water reactor (LWR) fuel in United States commercial power reactors. Current burnup limits vary between the fuel vendors but are at either 62 giga-watt days (energy produced) per metric ton of uranium (mass of uranium) GWd/MTU rod-average burnup or 70 GWd/MTU peak pellet burnup. These two limits are essentially the same given how LWR fuel is operated. PNNL has been tasked with providing technical assistance to the NRC related to the review and approval of licensing topical reports requesting burnup extension beyond the current limit. This report will provide the agency with expert technical assistance to enhance the staff's knowledge base of specific phenomena and damage mechanisms that are exhibited at high burnup and will support the agency's efforts to develop and review the required regulatory structure to support the licensing of high burnup fuels.

This report will provide background for burnup extension, the current regulatory structure, and previous guidance for burnup extension as it relates to in-reactor performance and dry storage and transportation of spent nuclear fuel. Next, an overview of burnup-related degradation phenomena will be provided. Changes to fuel performance safety analysis codes, methods, and design limits will be discussed in the context of burnup extension to a rod-average burnup of 62-85 GWd/MTU. This report will also identify any potential new damage mechanisms that are unique to this burnup level. A similar discussion will be provided for safety analysis methods for dry storage and transportation of high burnup spent nuclear fuel. Finally, a discussion of the current out-of-pile and in-pile data will be provided with special consideration to availability or lack of data to support the damage mechanisms and performance considerations previously identified.

Acknowledgments

This work was funded by the U.S. Nuclear Regulatory Commission under contract NRC-HQ-25-14-D-0001.

Acronyms and Abbreviations

AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CoC	Certificate of Compliance
CRUD	Chalk River Unknown Deposit (generic term for deposits on fuel cladding)
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	U.S. Department of Energy
DSS	Dry Storage System
EPMA	Electron Probe Micro-Analysis
GNF	Global Nuclear Fuels
HAC	Hypothetical Accident Conditions
IFA	Instrumented Fuel Assembly
IFBA	Integral Fuel Burnable Absorber
ISG	Interim Staff Guidance
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LTR	Licensing Topical Report
LTA	Lead Test Assembly
LWR	Light Water Reactor
MCPR	Margin to Critical Power Ratio
NCS	Normal Conditions of Storage
NCT	Normal Conditions of Transportation
NRC	U.S. Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
PIE	Post-Irradiation Examination
PNNL	Pacific Northwest National Laboratory
PWR	Pressurized Water Reactor
RIA	Reactivity-Initiated Accident
SAFDL	Specified Acceptable Fuel Design Limit
SNF	Spent Nuclear Fuel
SRP	Standard Review Plan

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1.0 Introduction

The U.S. Nuclear Regulatory Commission (NRC) is preparing for anticipated licensing applications requesting burnup extension for light water reactor (LWR) fuel in United States commercial power reactors. Current burnup limits vary between the fuel vendors but are at either 62 giga-watt days (energy produced) per metric ton of uranium (mass of uranium) (GWd/MTU) rod-average burnup or 70 GWd/MTU peak pellet burnup. These two limits are essentially the same given how LWR fuel is operated. Before the 1980's maximum burnup levels were around 30 GWd/MTU rod-average burnup. In the 1980's and 1990's the U.S. Department of Energy (DOE) sponsored several programs with all the nuclear fuel vendors in the U.S. to demonstrate improved fuel utilization of commercial nuclear fuel by sponsoring irradiation of lead test assemblies and the subsequent hot cell examinations (see for example (Smith G. , 1985) (Smith, Ruzauskas, Pirek, & Griffiths, 1993) (Barner, Cunningham, Freshley, & Lanning, 1990)). The result of these programs was NRC's approval of burnup limits of effectively 62 GWd/MTU rod-average burnup for all U.S. nuclear fuel vendors by 2000. Since this time, no further burnup extensions have been granted.

PNNL has been tasked with providing technical assistance to the NRC related to the review and approval of licensing topical reports requesting burnup extension beyond the current limit. This report will provide the agency with expert technical assistance to enhance the staff's knowledge base of specific phenomena and damage mechanisms that are exhibited at high burnup and will support the agency's efforts to develop and review the required regulatory structure to support the licensing of high burnup fuels.

This report will provide current state of the industry information on material properties and fuel performance considerations for high burnup fuels between 62 and 85 GWd/MTU rod-average burnup in operating reactor conditions, reactor design basis accident conditions, dry storage, and transportation. It is noted that to capture various phenomena that occur on a rod-average or pellet basis, it may be desirable to impose both rod-average and peak pellet burnup limits. To support the agency's efforts, this report will identify and discuss degradation and failure modes observed at high burnup including fuel performance characteristics of high burnup fuel that may not be addressed within existing regulatory documents (e.g., 10 CFR, regulatory guidance, standard review plans).

The scope of this report includes high burnup LWR fuel with Zr-alloy cladding (Zircaloy-2, Zircaloy-4, ZIRON, ZIRLO™, Optimized ZIRLO™, and M5®) and UO₂ fuel. This report does not cover cladding types that have not been approved by the NRC such as Cr-coated Zr-alloy cladding. This will also apply to fuel with various burnable absorbers including UO₂-Gd₂O₃, UO₂-Er₂O₃, and integral fuel burnable absorber (IFBA) rods (UO₂ with ZrB₂ coating). Changes to safety analysis codes, methods, and design limits for in-reactor performance will be discussed in the context of high burnup fuel for both normal conditions and accident conditions (Section 3.0). This section will also identify any potential new damage mechanisms that manifest at high burnup. The discussion of these damage mechanisms will also include discussion of existing or potential new data that could be used to develop or confirm design limits and performance relative to these design limits. Changes to safety analysis codes, methods, and design limits for wet storage, dry storage, and transportation of high burnup spent nuclear fuel will be discussed in Section 4.0. Finally, a discussion of the current out-of-pile and in-pile data will be provided in Section 5.0 with special consideration to availability or lack of data to support the damage mechanisms and performance considerations previously identified. This report does not discuss issues related to final fuel

disposal. It also does not discuss impacts on operations such as fuel management and operating limits and setpoints.

1.1 Background

Burnup is a measure of the energy derived from nuclear fuel. Higher burnup in fuel represents higher fuel utilization, but also results in more damage to the fuel and the cladding. Burnup is typically reported in terms of GWd/MTU. However, the energy released is not the primary cause of damage in the fuel and the cladding. The energy release results from the fission process, which produces around 200 MeV (3.2×10^{-11} J) per fission. Each fission results in the destruction of one uranium atom and its replacement by two or more fission products. This process results in the formation of voids and interstitial products in the ceramic UO_2 matrix. Additionally, the fission process results in the production of neutrons. Fast neutrons (> 1 MeV) cause further lattice damage in both the ceramic UO_2 fuel and metallic Zr-alloy cladding. This damage leads to several phenomena that impact the thermal-mechanical fuel performance. Fast neutron fluence in a LWR is proportional to burnup with around 1.67×10^{24} n/m² for every 1 GWd/MTU. This results in a fast neutron fluence of 1×10^{26} n/m² after 60 GWd/MTU.

For LWRs the cladding has historically been fabricated from zirconium alloys. For boiling water reactors (BWRs) the alloy Zircaloy-2 was used. For pressurized water reactors (PWRs) the alloy Zircaloy-4 has been used. As demand for higher burnup levels (up to 62 GWd/MTU) came for LWR fuels, in-reactor cladding corrosion became a problem. To reduce the in-reactor corrosion and maintain or improve the creep properties of the cladding, the nuclear fuel vendors have developed proprietary, Zr-based cladding alloys that have mostly replaced the use of traditional Zircaloy alloys. Westinghouse now uses the alloys ZIRLO™ and Optimized ZIRLO™ for their PWR fuel, while retaining Zircaloy-2 for BWR fuel. Framatome uses M5® for their PWR fuel, while also retaining Zircaloy-2 for BWR fuel. There may be limited PWR application of Zircaloy-4 by Westinghouse and Framatome. Global Nuclear Fuels (GNF) only supplies BWR fuel and has recently received approval for ZIRON cladding.

General design criteria for nuclear power plants are defined in 10 CFR 50 Appendix A (US Nuclear Regulatory Commission, 2017). The specific damage and failure mechanisms for fuel that have historically been identified for LWR fuel are identified in the Standard Review Plan Section 4.2 (US Nuclear Regulatory Commission, 2007) and discussed in greater detail in Section 3.3. In general, safety analysis is performed prior to operation to show:

- Rods will not fail during normal operation and anticipated operational occurrences (AOOs)
- Rods may fail during a design basis accident. If rods fail during a design basis accident, the number of failed rods should not be underestimated for dose considerations and failure should not result in a loss of coolable geometry.

1.1.1 Normal Operation and Anticipated Operational Occurrences

Although fuel rods are static components, the cladding is exposed to tensile and compressive stresses and exhibits strain in several directions. Early in life, the fuel-cladding gap is open, and the external pressure is much greater than the internal pressure. Because of this, 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 creepdown, the fuel-cladding gap closes, and continued pellet swelling causes the

cladding to strain outward in the hoop direction. Later in life, if sufficient amounts of fission gasses 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. In addition to these deformations in the hoop direction, the cladding exhibits a hexagonal crystal structure and is highly textured, such that radiation causes growth in the axial direction. Additionally, when the pellet-cladding gap is closed, pellet swelling in the axial direction can result in further cladding strain in the axial direction. Finally, through the reaction of Zr with water, a corrosion layer of ZrO_2 is observed to build up on the cladding outer surface. This corrosion layer grows at the expense of the cladding thickness. Due to density differences in ZrO_2 and Zr, the thickness of the ZrO_2 is 1.56 times greater than the thickness of the cladding that is reduced. This difference is known as the Pilling-Bedworth ratio (Pilling & Bedworth, 1923).

The reactor conditions that LWR cladding have been exposed to under normal operations are as follows:

Boiling Water Reactor (BWR)

- Liquid water from 530°F (277°C) to 550°F (288°C) at 1035 psi (7.1 MPa) and steam at 550°F (288°C) and 1035 psi (7.1 MPa)
- Coolant mass flux of $\sim 1.05 \times 10^6$ lb/ft²-hr (1427 kg/m²-s)
- Fast neutron flux 1×10^{18} n/m²-s
- Time in core of 1500 to 2000 days
- Rod-average burnup of 62 GWd/MTU

Pressurized Water Reactor (PWR)

- Liquid water 550°F (288°C) to 610°F (321°C) at 2250 psi (15.5 MPa)
- Coolant mass flux of $\sim 2.55 \times 10^6$ lb/ft²-hr (3466 kg/m²-s)
- Fast neutron flux 1×10^{18} n/m²-s
- Time in core of 1500 to 2000 days
- Rod-average burnup of 62 GWd/MTU.

The cladding conditions during AOOs are not significantly different than those during normal operation and typically 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

As mentioned before, design basis accidents have been identified for LWRs and during these events failure of the cladding is permitted, but the number of failed rods should not be underestimated, and the failure of rods should not impact the coolability of the fuel assembly. The main design basis accidents of interest to the fuel design review are: reactivity-initiated accident (RIA) and loss-of-coolant accident (LOCA). The conditions and fuel damage anticipated are described below.

Reactivity-Initiated Accident (RIA)

This accident is caused by a rapid, uncontrolled removal of a control rod or control blade from the core that results in an extreme increase in power in nearby fuel rods (1000 times increase) over a very short time (~20 ms) that then goes back to zero power. This event results in thermal expansion of the pellet, which can contact the cladding and causes relatively large (1-5%) hoop strain in the cladding at relatively low temperature (<700°C). This pellet-clad mechanical interaction can cause cladding failure and, if extreme enough, can lead to violent expulsion of the fuel from the cladding which can result in a loss of coolable geometry or a pressure pulse that can damage the reactor vessel.

Loss-Of-Coolant Accident (LOCA)

This accident is a loss-of-coolant in the core. In the case of a large pipe break, the accident can involve rapid depressurization of the reactor core and complete loss of water to the core. In the case of a small pipe break, the accident may be characterized by a slower depressurization and partial loss of water to the core. Although the fission process is stopped by negative void reactivity feedback, this loss of active cooling leads to heating of fuel rods from decay heat. Ballooning and burst of fuel rods are observed between 800-1000°C and high temperature oxidation of cladding with steam, an exothermic reaction which creates additional heat, is observed between 1000°C and 1200°C. At some point during the event, the emergency core cooling system will reflood the reactor with water, resulting in potential rapid cooling of the fuel rods by water quench. Numerous mechanisms for fuel cladding failure exist in the accident, including ballooning and burst where fuel may be ejected from the fuel rods and high temperature corrosion could embrittle the cladding, leading to fuel fracture and a loss of coolable geometry during the reflood phase.

1.1.3 Beyond Design Basis Accident Conditions

The Fukushima accident would be considered a beyond design basis accident. In this event there was a long-term loss of offsite power and no onsite generating capacity, leading to an inability to remove decay heat from the shut-down reactor core. After an extended period, the water in the core boiled off and the cladding reacted with the steam to produce hydrogen. The hydrogen was not properly vented from the reactor building and after a critical concentration of hydrogen accumulated it caused an explosion. In the U.S. there are engineered systems employed to prevent such explosions. Currently the U.S. has no regulations related to fuel performance and qualification during events and accidents classified as “beyond design basis.”¹ However, work is currently being done with the goal of improving the performance of the fuel assemblies at temperatures above 1200°C (Brachet, et al., 2018) (Oelrick, Xu, Lahoda, & Deck, 2018) which would currently represent performance beyond design basis.

¹ Note that NRC has various requirements for beyond design basis accidents, including the station blackout rule (10 CFR 50.63), the anticipated transient without scram rule (10 CFR 50.62), and requirements for maintaining or restoring core and spent fuel pool cooling and containment integrity in the event of large explosions or fires (10 CFR 50.54 (hh)(2), also known as B.5.b). NRC has also published a final rule (US Nuclear Regulatory Commission, 2019) governing various aspects of beyond design basis accidents that originated as part of the post-Fukushima lessons learned activities. However, none of these rules and regulations establish specific requirements for fuel performance or qualification for beyond design basis accidents.

1.2 Previous Guidance

The NRC has a robust system for review and approval of new safety analysis codes and methods. For a fuel vendor to be able to perform cycle-specific safety analyses on a new or existing fuel assembly design where operation deviates from limits applied to their currently approved methodologies, that vendor would typically prepare and submit new licensing topical reports (LTRs) to the NRC to describe the codes and methods that would be used to perform these analyses. The in-reactor safety analysis which is part of the core design process, is performed prior to each cycle. During this process, if an analysis indicates that any of the thermal-mechanical Specified Acceptable Fuel Design Limits (SAFDLs) are exceeded, a core redesign will be triggered. This process will continue until an adequate core loading pattern and operating plan has been determined.

The safety analyses for storage and transportation of fresh and spent nuclear fuel (SNF) is somewhat different than the in-reactor performance. The storage and transportation have not been historically considered as a part of the fuel design process. There are currently no in-reactor operating restrictions that are in place because of SNF considerations. However, the peak cladding temperature and average-rod hoop stresses during drying operations are limited via guidance. These limits may need to be revised depending on the end-of-life condition. Additionally, the individual rod power histories are not available for the analysis of SNF during drying, loading, storage, or transportation. Therefore, the safety limits for storage and transportation of SNF must be developed for the most limiting fuel rods at the maximum expected burnup.

The following sub-sections will provide an overview of the documents that provide the regulatory framework for the safety analyses regarding the irradiation, dry storage, and transportation of LWR fuel. Section 1.2.1 briefly discusses the documents for fuel irradiation and Section 1.2.2 discusses the documents for SNF storage and transportation. Recent NRC staff insight relating to higher burnup are highlighted.

1.2.1 Irradiation of Fuel to High Burnup

The regulations related to in-reactor fuel performance are provided in 10 CFR Part 50 –Domestic Licensing of Production and Utilization Facilities (US Nuclear Regulatory Commission, 2017). To assist staff in the review of licensing submittals, the NRC provides a standard review plan, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” – NUREG-0800. Specifically, Chapter 4 – Reactor, Section 4.2 – Fuel System Design (US Nuclear Regulatory Commission, 2007) provides the SAFDLs for normal operation, AOO, and accident conditions. The AOO and accident analysis is further described in Chapter 15 – Transient and Accident Analysis (US Nuclear Regulatory Commission, 2007).

Recently, in 2015 NRC staff identified issues with high burnup fuel (Clifford, 2015). Clifford (2015) identified the following challenges of high burnup and extended in-reactor service:

- Progressive changes in fuel pellet microstructure and properties
- Higher decay heat loads
- Higher fission gas release and rod internal pressures

- Progressive changes in fuel cladding microstructure and properties
 - Cladding corrosion and hydrogen uptake
- Dimensional changes in fuel rod and assembly components due to irradiation-induced growth, creep, and corrosion
 - Spacer spring relaxation
 - BWR channel distortion.

Clifford (2015) also identified pellet and cladding characteristics that evolve with burnup as follows:

- Evolving Pellet Characteristics
 - Pellet cracking due to thermal stresses
 - Accumulation of fission products
 - Decrease in fuel thermal conductivity
 - Growth of porous rim structure and fuel-clad bonding
- Evolving Cladding Characteristics
 - Irradiation damage (point defects and dislocation loops) increases yield strength and decreases ductility
 - Irradiation-induced free growth and irradiation-assisted creep alter as-fabricated dimensions
 - Water side corrosion results in a progressive ZrO₂ layer
 - Hydrogen uptake in the base metal leads to the formation of brittle zirconium hydrides

Finally, this Clifford (2015) identified recent concerns that may require changes to the regulatory framework.

- Cladding hydrogen decreases ductility and may leave the cladding with no ductility during postulated LOCAs
- Hydrides decrease cladding ductility and the ability of the cladding to withstand pellet thermal expansion during over power AOOs and accidents

These phenomena and their impact on in-reactor safety analysis will be discussed in greater detail in Section 3.0.

Previous work has been done by NEI and EPRI to establish benefits and challenges with increasing fuel enrichment and burnup for LWRs (Pimentel & Smith, 2019). Although Pimentel & Smith (2019) primarily focuses on economic issues, it does identify fuel fragmentation that has been observed at high burnup as a technical issue. This issue is discussed in this report in Section 3.3.4.

1.2.2 Storage and Transportation of Spent Nuclear Fuel

The regulations related to wet storage of SNF are provided in 10 CFR Part 50 Domestic Licensing of Production and Utilization Facilities (US Nuclear Regulatory Commission, 2017) as General Design

Criteria (GDC) 61 “Fuel Storage and Handling and Radioactivity Control”. Specifically, GDC 61 requires (1) periodic inspections; (2) suitable radiation shielding; (3) appropriate containment, confinement, and filtering systems; (4) residual heat removal capability consistent with its importance to safety; and (5) prevention of significant reduction in fuel storage inventory under accident conditions. To augment those requirements, the spent fuel pool design basis is also covered by GDC 2, “Design Bases for Protection Against Natural Phenomena”; GDC 4, “Environmental and Dynamic Effects Design Bases”; and GDC 63, “Monitoring Fuel and Waste Storage.”

To assist staff in the review of licensing submittals, the NRC has provided Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis” (US Nuclear Regulatory Commission, 2007). This document discusses considerations that should be made to account for high burnup fuel.

The regulations related to dry storage of SNF are provided in 10 CFR Part 72 – Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste (US Nuclear Regulatory Commission, 2015). The regulations related to transportation of SNF are provided in 10 CFR Part 71 - Packaging and Transportation of Radioactive Material (US Nuclear Regulatory Commission, 2015). The standard format and content for Part 71 applications for approval of packages for radioactive material is given in Regulatory Guide 7.9 (US Nuclear Regulatory Commission, 2005).

To assist staff in the review of licensing submittals, the NRC provides several standard review plans, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility” – NUREG-1536 Rev. 1 (US Nuclear Regulatory Commission, 2010), “Standard Review Plan for Spent Fuel Dry Storage Facilities” – NUREG-1567 (US Nuclear Regulatory Commission, 2000) and “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel” – NUREG-1617 (US Nuclear Regulatory Commission, 2000). NRC staff guidance on performing these reviews has been augmented by Interim Staff Guidance 11 Rev 0, 1, 2, and 3 to provide additional details and technical basis for the storage of spent fuel including assemblies with average burnups exceeding 45 GWd/MTU. Other interim staff guidance has also been provided including ISG-19.

NRC staff have recently provided a draft NUREG that is out for comment (NUREG-2224) (Ahn, et al., 2018) regarding dry storage and transportation of high burnup SNF. NUREG-2224 considers burnups >45 GWd/MTU assembly average, and considers the cladding/material characteristics at currently-discharged burnups. It does not address the higher burnups discussed in this current report. Additionally, NRC staff presented their thoughts on the management of high burnup spent fuel (Torres, 2018). Regarding storage and transportation of SNF, these documents discussed the roles of cladding creep, hydride reorientation, and cladding fatigue lifetime. These documents also provide guidance for dry storage and transportation for less than 20 years and dry storage for greater than 20 years. Regarding operability and safety significance of aging mechanisms of the fuel cladding/assembly hardware on the performance of the fuel for dry storage periods up to 60 years, NRC has issued NUREG-2214 (US Nuclear Regulatory Commission, 2019) that provides the technical basis for these issues.

The implications of these documents and their guidance on approving storage and transportation of SNF will be discussed in greater detail in Section 4.0

2.0 High Burnup Phenomena

There are several changes that take place in an LWR fuel rod as burnup progresses. Changes occur in the fuel pellets, the fuel/cladding gap, and the cladding. Within the fuel, fission is taking place, which leads to the destruction of uranium atoms and the replacement of these atoms with two or more fission products. This leads to swelling of the fuel pellets and the release of xenon and krypton gas. Neutron radiation, particularly fast neutron irradiation ($E > 1$ MeV), also causes defects in the fuel that along with the production of fission products results in degradation of the thermal conductivity of the pellet. The cladding is damaged by fast neutron radiation, which is proportional to burnup in the pellet. Fast neutron irradiation leads to axial growth in the cladding, increase in the cladding strength and creep rate, and decrease in the cladding ductility. The cladding metal/water reaction continues with time in reactor and leads to a thicker oxide layer and thinner cladding with increasing burnup. The fuel/cladding gap size is impacted by deformation of the fuel and the cladding. The composition of the plenum gas is impacted by the release of xenon and krypton gas from the fuel pellets.

The following sections will discuss the changes to the fuel pellets, fuel/cladding gap, and the cladding that occurs with burnup and will specifically focus on changes that occur beyond a rod-average burnup of 62 GWd/MTU up to 85 GWd/MTU.

2.1 Fuel

The fraction of fission gas that is released from the fuel pellets generally increases with increasing burnup. For normal LWR rods, a release fraction of about 1% per 10 GWd/MTU beyond 40 GWd/MTU is observed for very low power rods with another 2-3% for higher power rods. Beyond about 60 GWd/MTU, the fraction of gas that is released begins to increase at an exponential rate. See Figure 2.1 where by 80 GWd/MTU, a release of 15% is observed and 22% by 100 GWd/MTU (Manzel & Walker, 2002). Depending on operating conditions such as power history, the fission gas release at high burnup could be even greater than this. In addition to the fission gas release fraction increasing, the absolute quantity of fission gas increases with burnup. Existing fission gas release models likely predict the increase in gas production but may not predict this exponential release of fission gas release beyond 60 GWd/MTU although may provide reasonable predictions up to 62 GWd/MTU. Updated models and assessment data should be provided, as the increased fission gas release can lead to decreased gap conductivity and increased rod internal pressure. It will also result in a higher source term for accident analyses, so if fission gas release fractions are assumed for these analyses, the bases for these assumptions should be re-evaluated for high burnup (62-85 GWd/MTU). Additionally, as burnup progresses, the Xe/Kr ratio that is produced changes as more of the fissions come from plutonium than from uranium. For uranium fission, the Xe/Kr ratio is 5.67 while for plutonium it is 16 (White, et al., 2001). This change in ratio does not have a strong safety significance but should be considered in a high burnup fission gas release model.

Finally, another use of fission gas release models is in the assessment of radioactive source term for failed fuel. Currently, NRC publishes tables with maximum expected source terms in Regulatory Guide 1.183 (US Nuclear Regulatory Commission, 2000). These tables are specifically applicable to a rod-average burnup of 62 GWd/MTU. Upper bound predictions using a stable and radioactive fission gas release model that has been assessed to higher burnup will have to be used to extend these tables to higher burnup.

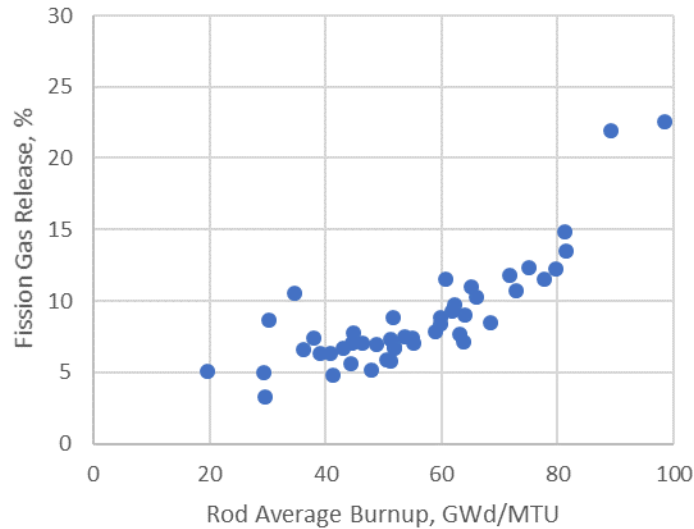


Figure 2.1. Typical fission gas release at high burnup (Manzel & Walker, 2002)

As burnup progresses beyond 40 GWd/MTU, the outer radius of the pellets begins to exhibit the formation of what is called the “high burnup rim”. The high burnup rim can be characterized in terms of submicron grains with low angle grain boundaries and high porosity (Manzel & Walter, 2000) (Spino, Vennix, & Coquerelle, 1996) (Une, Mogita, Shiratori, & Hayashi, 2001). Figure 2.2 shows the observed thickness of the high burnup rim.

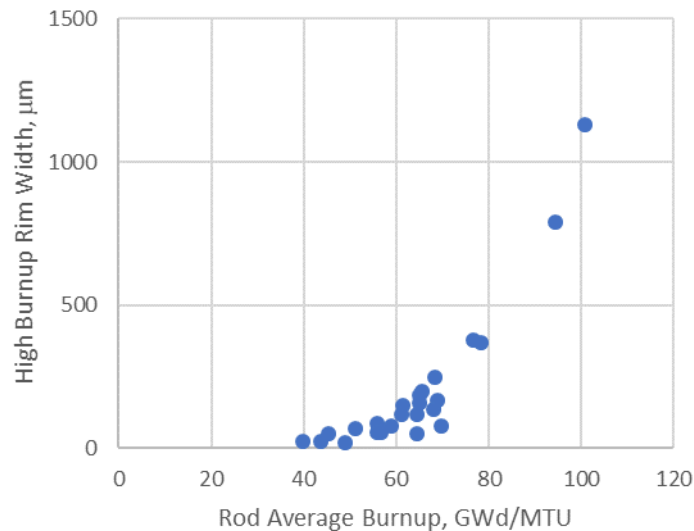


Figure 2.2. Typical high burnup rim thickness (Manzel & Walker, 2002) (Typical pellet radius is 5000 μm)

Although there is no direct performance impact from the high burnup rim, the increased porosity in this area holds a significant quantity of fission gas that could be released during any design basis event and the fuel-clad bonding layer may impact cladding deformation. This could increase the potential radioactive source term for failed fuel during these accidents. Additionally, drying operations on spent fuel have the potential to release this gas stored in the high burnup rim through non-gross rupture in the cladding. If

fission gas release fractions are assumed for these analyses, the bases for these assumptions should be re-evaluated for high burnup (62-85 GWd/MTU).

The submicron grain size in the high burnup rim could lead to increased pellet fragmentation following ballooning and burst during a LOCA. Recent tests at Halden and Studsvik (Flanagan, Askeljung, & Puranen, 2013) (Oberlander & Wiesenack, 2014) (Raynaud P. , 2012) have revealed some pellet fragmentation and relocation out of the burst opening for rods with burnup above 60 GWd/MTU and considerably more fragment dispersal above 80 GWd/MTU. This pellet release could challenge coolability limits following LOCA. This could also become an issue for RIA following cladding failure of high burnup fuel. The pellet rim should also be considered in determining the amount of material that may be released through non-gross ruptures in undamaged fuel during drying, storage and transport operations.

Thermal conductivity of the UO_2 pellets is significantly degraded as burnup progresses. The thermal conductivity of UO_2 is already quite low. As a ceramic material, the heat transfer is primarily by phonon heat transfer or lattice vibrations. Since there are no free electrons, the electronic contribution only becomes significant at very high temperature ($> 3000\text{K}$). As burnup progresses, the formation of voids and interstitials disrupt the UO_2 crystal lattice and reduce its ability to transfer heat. In addition, the radiation damage from fast neutrons causes further damage to the lattice by creating dislocations. All this lattice damage results in a reduction in the UO_2 thermal conductivity. The impact of reduced thermal conductivity is increased fuel temperature. Increased fuel temperature drives many of the safety concerns with in-reactor fuel performance.

The melting temperature of UO_2 is observed to decrease with increasing burnup. This decrease in melting temperature is only 1 to 5°C per GWd/MTU, and this represents a decrease in safety margin for in-reactor fuel performance.

UO_2 fuel has been observed to swell with increasing burnup. This swelling is due to the accumulation of solid fission products in the UO_2 matrix. After the initial densification that occurs in the pellet early in life the swelling proceeds at a linear rate with burnup. At very high burnup above 85 GWd/MTU, the linear swelling rate has been observed to increase beyond the rate that was observed at lower burnups.

Additionally, during long term dry storage, fuel swelling is observed caused by alpha-decay of the radioactive nuclides in spent fuel, where the alpha particles knockon atoms in the UO_2 lattice. The resulting recoil nuclei result in displacement cascades and in the creation of Frenkel pairs, defects that ultimately result in lattice swelling (Raynaud & Einziger, 2015). This swelling should be considered during the long term storage of SNF.

The production of power is not uniform across the radius of the pellet. Early in life the radial power profile is relatively uniform with a small peak on the outer edge. This preference for production on the outer edge of the pellet is due to the neutron self-shielding effect. As burnup progresses, the slight edge peak of the radial power profile becomes more and more pronounced due to a combination of the self-shielding effect and the preferential build-in of plutonium from ^{238}U neutron capture. By high burnup, the power at the edge of the pellet can be a factor of two to three greater than average power of the pellet. The edge peaked power also leads to an edge peaked burnup distribution. The accurate prediction of this power profile impacts the prediction of fuel temperature, which impacts other predictions such as fission gas release.

2.2 Fuel/Cladding Gap

Beyond a burnup level of around 20 GWd/MTU, the fuel/cladding gap is typically closed due to a combination of pellet swelling and cladding creepdown. For this reason, operation to higher burnup will not likely impact the heat transfer characteristics of the fuel/cladding gap if the gas gap pressure remains below the system pressure. As burnup progresses a fuel/clad bonding layer also begins to form. This bonding layer is primarily observed as it makes it difficult to remove fuel from the cladding during hot cell examinations on high burnup fuel. However, this bonding layer is poorly characterized and is not typically used in any modeling of high burnup fuel.

As burnup progresses more fission gas is produced in the pellet, such that even if the fission gas release fraction is constant, more fission gas will accumulate in the fuel-cladding gap. Because of this increase in fission gas and because there is larger fission gas release fraction at high burnup as noted in Section 2.1, and the fuel-cladding gap is closed, the rod internal pressure will begin to increase and at some burnup level, it is expected to exceed the external system pressure. As the rod internal pressure continues to increase, there is the potential that the fuel/cladding gap could reopen if the cladding creep-out rate were to exceed the fuel pellet swelling rate. This situation is undesirable as gap reopening would increase the fuel temperature, which could lead to a cycle of more fission gas release, lower gap conductance, and higher fuel temperature. There is a SAFDL (Section 3.3.2.5) that discusses this, and this SAFDL will become more limiting at high burnup.

2.3 Cladding

The cladding is impacted by burnup through two mechanisms. These are fast neutron fluence and long-term exposure to the reactor coolant water.

2.3.1 Impact of Fast Neutron Fluence

Fast neutron fluence is the primary cause of damage in the cladding at high burnup. Fast neutron fluence in a LWR is proportional to burnup with around 1.67×10^{24} n/m² for every 1 GWd/MTU. This results in a fast neutron fluence of 1×10^{26} n/m² after 60 GWd/MTU. For high burnup between 62 and 85 GWd/MTU, this corresponds to fast neutron fluence levels of 1×10^{26} n/m² to 1.5×10^{26} n/m².

Some of the mechanical properties of Zircaloy and other Zr-alloy claddings are strongly impacted by fast neutron fluence. The yield stress and ultimate tensile strength rapidly increases to about 1.5 to 2 times the unirradiated value. This strengthening occurs rather early in life by about 2×10^{25} n/m² (~ 15 GWd/MTU) with only moderate further increase until about 8×10^{25} n/m² (~50 GWd/MTU). After this, the strengthening effect appears to saturate with no further increase with fluence. In contrast with the increase in strength, Zr-alloy cladding exhibits a marked decrease in ductility. This decrease in ductility is from a combination of irradiation hardening and hydrogen pickup discussed later. The elastic modulus is only weakly impacted by fast neutron fluence. In general, the unirradiated modulus may reliably be used to model the elastic deformation of irradiated Zr-alloy cladding at all fluence levels.

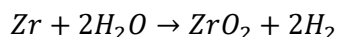
Zr is a hexagonal crystal material and the formation of cladding tubes results in a large degree of texture in the cladding tubes (i.e., preferential crystal orientation). This texture causes Zr-alloy cladding tubes to grow in the axial direction with increasing fast fluence. This axial growth is typically modeled using a

power law. Some proprietary data indicates that axial growth in some alloys may increase beyond what is expected at high fluence.

Cladding creep is impacted by neutron irradiation and is typically modeled as a combination of thermal creep and irradiation creep. The irradiation creep rate does not appear to change with increasing fast neutron fluence. There does appear to be some hardening that occurs that reduces the thermal creep rate somewhat with increasing fluence.

2.3.2 Impact of Exposure to Reactor Cooling Water

A secondary damage mechanism for the cladding caused by exposure to the primary coolant water is the metal-water reaction and subsequent hydrogen pickup. The metal-water reaction is given by,



This reaction results in the conversion of the outer surface of the cladding tubes to zirconium oxide, which effectively thins the cladding. Additionally, this reaction results in free hydrogen. Some of this hydrogen is picked up by the cladding and precipitates out in the cladding as brittle zirconium hydrides.

With increasing burnup, the rate of oxide thickness growth may deviate from the previously observed relatively linear rate. Operation to high burnup may challenge oxide thickness limits that are set by the vendor. Additionally, in BWR cladding, the hydrogen pickup fraction has been observed to increase at high burnup due to the dissolution of second phase precipitate particles (Geelhood & Beyer, 2011). It is possible that something similar may happen in PWR cladding at very high burnup. Additional hydrogen content in the cladding can lead to brittle behavior of the cladding. Safety analyses require that the cladding retain some ductility, so excessive cladding hydrogen content could challenge these limits.

3.0 Changes to In-Reactor Safety Analysis Codes, Methods, and Limits for High Burnup Fuel

For a fuel vendor to be able to perform cycle-specific safety analyses on a new or existing fuel assembly design where operation deviates from limits applied to their currently approved methodologies, that vendor would typically prepare and submit new LTRs to the NRC to describe the codes and methods that would be used to perform these analyses. Alternatively, the fuel vendor could prepare and submit supplements to existing LTRs that state changes to the codes and methods that would be used to perform safety analyses for operation over an expanded range.

In the case of burnup extension, it is likely that a new fuel assembly design with new materials will be introduced. Rather models and methods will be changed or justified for use in the extended burnup range. In this case, likely the second approach of submitting supplements to existing LTRs will be used. The content of these LTR supplements should describe the material property correlations that will be used over the extended burnup range along with data to justify the use of each correlation. Even in the case where an existing correlation will be used for the fuel or the cladding, some justification, preferably citing data that the use of the correlation is appropriate, should be provided. It is the responsibility of the applicant to propose SAFDLs for their analyses and justify those limits. Because of this, these LTR supplements should also state these limits over the entire burnup range and provide relevant data that demonstrate that these limits will adequately protect the fuel assembly up to the requested burnup level. Finally, if any changes are made to the analytical methodology including the magnitude of the model uncertainties used to perform the safety analyses at high burnup, these changes should also be described in the LTR supplement or similar licensing document.

This general approach of licensing fuel thermal-mechanical codes and methods can be used as a model for the licensing of operation to high burnup. The same review and approval will be required of any licensing requests in the following three areas:

- Material property correlations to be used in codes for high burnup fuel and cladding
- Any changes to SAFDLs at high burnup
- Any changes to existing methodology for high burnup fuel including new uncertainties for high burnup.

This section is intended as a guide for NRC staff as they perform a review of an LTR, LTR supplement, or license amendment request related to the approval of rod-average burnup greater than 62 GWd/MTU. Section 3.1 provides a list of fuel material property correlations that are typically needed to adequately model fuel system response based on development and qualification of NRC's independent fuel performance code, FRAPCON (Geelhood K. , Luscher, Raynaud, & Porter, 2015) and based on all previously approved thermal-mechanical codes. Section 3.2 provides a list of cladding material property correlations that are typically needed to adequately model fuel system response based on development and qualification of NRC's independent fuel performance codes, FRAPCON (Geelhood K. , Luscher, Raynaud, & Porter, 2015) and FRAPTRAN (Geelhood K. , Luscher, Cuta, & Porter, 2016), and based on all previously approved thermal-mechanical codes. These sections also identify data that are typically used to develop and justify these correlations.

Section 3.3 discusses SAFDL limits in areas that are identified in Standard Review Plan Section 4.2 (US Nuclear Regulatory Commission, 2007). This section also identifies data that are typically used to develop and justify these limits. This section will also identify potential new damage mechanisms that should be considered specifically for high burnup fuel and cladding along with data that could be used to justify proposed limits. Section 3.4 discusses potential changes to existing codes and methodologies that may be enacted to perform safety analyses for high burnup fuel.

3.1 Fuel Material Property Correlations

The following fuel material properties are typically needed to perform fuel thermal-mechanical analysis of UO_2 nuclear fuel with Zr-alloy cladding under normal conditions and AOOs:

- Thermal conductivity
- Thermal expansion
- Emissivity
- Enthalpy and specific heat
- Melting temperature
- Densification
- Swelling
- Fission gas release
- Radial power profile
- Fuel radial relocation
- High burnup rim formation
- Decay heat

The fuel material properties used in FRAPCON are described in a materials property handbook (Luscher, Geelhood, & Porter, 2015) and are generally applicable to high burnup. Table 3.1 provides a summary of the tests that could be performed to quantify the material properties discussed below.

3.1.1 Thermal Conductivity

As discussed in Section 2.1 the thermal conductivity of UO_2 fuel is known to decrease with increasing burnup (Lanning, Beyer, & Geelhood, 2005) (Ohira & Itagaki, 1997). Current thermal conductivity models in FRAPCON (Geelhood K. , Luscher, Raynaud, & Porter, 2015) are based on ex-reactor measurements on pellets and disks up to 90 GWd/MTU and also comparison to Halden temperature measurements up to 80 GWd/MTU.

A fuel thermal conductivity model based on laser flash diffusivity data from high burnup pellets and comparisons to high burnup in-reactor temperature measurements should be acceptable up to a rod average burnup of 85 GWd/MTU.

3.1.2 Thermal Expansion

Fuel thermal expansion is not expected to change significantly with irradiation based on the currently available data. Thermal expansion is caused by crystal lattice expansion (Van Vlak, 1989) and does not change much with the introduction of dislocations from fast neutron irradiation. The introduction of solid and gaseous fission products is accounted for separately by the swelling correlations discussed in Section 3.1.7. No change in thermal expansion with irradiation is used in FRAPCON (Luscher, Geelhood, & Porter, 2015). Thermal expansion data as a function of temperature from unirradiated samples have typically been used to develop cladding thermal expansion correlations.

A fuel thermal expansion model based on unirradiated fuel thermal expansion data is acceptable for use up to 85 GWd/MTU.

3.1.3 Emissivity

Fuel emissivity is used to calculate the portion of the gap heat transfer due to radiative heat transfer. Although at reactor temperatures, this contribution is small, it can be more significant at high temperature during accident conditions. The emissivity is impacted by the surface condition of the fuel. No burnup dependence has been observed (Luscher, Geelhood, & Porter, 2015). Additionally, radiative heat transfer is most important early in life when the fuel-cladding gap is open at burnup levels below to 20-30 GWd/MTU. At high burnup, for normal operations and AOOs, analysis will be done to show the fuel-cladding gap is closed and in that cases heat transfer is dominated by contact conduction (Geelhood K. , et al., 2009).

A fuel emissivity model based on unirradiated fuel emissivity data is acceptable for use up to 85 GWd/MTU.

3.1.4 Enthalpy and Specific Heat

Fuel enthalpy and specific heat are not expected to change significantly with irradiation based on the currently available data. Specific heat of a material is primarily dependent on the composition and the crystal structure (Gaskell, 1995) and has not been observed to change much with the introduction of dislocations from fast neutron irradiation. No burnup dependence has been observed. No change in enthalpy or specific heat with irradiation is used in FRAPCON (Luscher, Geelhood, & Porter, 2015). Enthalpy and/or specific heat data as a function of temperature from unirradiated samples have typically been used to develop cladding enthalpy and specific heat correlations.

A fuel enthalpy and specific heat model based on unirradiated fuel enthalpy and specific heat data is acceptable for use up to 85 GWd/MTU.

3.1.5 Melting Temperature

Fuel melting temperature in UO_2 displays a linear decrease with burnup. This decrease is based on data from unirradiated and irradiated UO_2 up to a burnup level of 100 GWd/MTU (Popov, Carbajo, Ivanov, & Yoder, 2000).

A fuel melting temperature model based on irradiated fuel melting temperature data should be acceptable up to 85 GWd/MTU.

3.1.6 Densification

Densification is a phenomenon that occurs early in life. Fuel typically reaches its maximum densification before 10 GWd/MTU (Rolstad, 1974). Modern fuel is fabricated with high theoretical density and hence densification is limited in these pellets. There has been no observation of continuing densification at higher burnup. No model updates are anticipated for densification in fuel above a rod-average burnup of 65 GWd/MTU.

3.1.7 Swelling

Once densification has occurred, fuel solid swelling is observed as a linear increase in volume with burnup. Halden has measured swelling rates from several instrumented fuel assemblies (Luscher, Geelhood, & Porter, 2015). From these tests a rate between 0.4 and 0.7 % $\Delta V/V$ per 10 GWd/MTU has been observed below 80 GWd/MTU. Above 80 GWd/MTU, a rate between 0.7 and 1.1 % $\Delta V/V$ per 10 GWd/MTU has been observed. Therefore, as rod-average burnup progresses past 80 GWd/MTU, the swelling rate should be increased (Luscher, Geelhood, & Porter, 2015).

A fuel solid swelling model based on swelling rate data should be acceptable up to 85 GWd/MTU.

In addition to solid swelling, the build-up of gaseous fission products can also cause pellet swelling that leads to cladding deformation. This gaseous swelling, as opposed to the solid swelling, does not increase with burnup, but rather provides a source of pellet swelling that is activated at high temperature, typically during a power ramp. Gaseous swelling has resulted in greater-than-expected cladding strains during power ramps above 40-50 GWd/MTU (Kallstrom, 2005) when the fuel temperature is between 950 and 1850°C (Mogensen, Walker, Ray, & Coquerelle, 1985). Although previous codes and methods have been approved without explicit gaseous swelling models to 62 GWd/MTU which is above the burnup level where gaseous swelling has been observed to impact cladding hoop strain during a power ramp, in many cases, uncertainties and conservatism were judged to cover this. For approval of fuel above 62 GWd/MTU, gaseous swelling should be considered, particularly for cladding strain during power ramps.

A gaseous swelling model that has been assessed on ramp test data from high burnup rods should be acceptable up to 85 GWd/MTU. Currently only limited ramp test data exist. (Wesley, Mori, & Inoue, 1994) (Kallstrom, 2005)

3.1.8 Fission Gas Release

As discussed in Section 2.1, as burnup proceeds beyond around 60 GWd/MTU, the fission gas release in low to moderate powered fuel rods begins to deviate from a rate of about 1% per 10 GWd/MTU to an exponential rate where the total release at 80 GWd/MTU is around 15% and 25% by 100 GWd/MTU (Manzel & Walker, 2002). Fission gas release is a complex interplay of temperature, fission rate, and burnup, and most models only use such generalizations above as the athermal term and add fission gas released due to higher power and temperature operation on to this term. Nevertheless, fission gas release models that have not been assessed against data at high burnup will likely underpredict fission gas release, which will lead to an underprediction of rod internal pressure and possibly an underprediction of fuel temperature. These two predictions are necessary to perform safety analyses on fuel.

A fission gas release model should be validated against data from high burnup fuel rods, and in particular fuel rods taken from limiting power and temperature locations, as these are the rods that will drive limits to rod internal pressure and fuel temperature (Geelhood & Luscher, 2015). When deriving uncertainties for a high burnup fission gas release model, it is recommended that an uncertainty range based only on high burnup data be established and used when doing safety analyses at high burnup. Because there is a large quantity of fission gas release data at low to moderate burnup, and a smaller database at high burnup, it is important that the large database at low burnup not impact the uncertainty due to larger scatter and a lower number of measurements at high burnup. A high burnup fission gas release model should also account for the change in Xe/Kr ratio that occurs as burnup progresses.

To use a fission gas release model to update source term tables for short-lived radionuclides, in-pile sweep tests of high burnup fuel would be necessary.

A fission gas release model with associated uncertainties derived from comparison to high burnup fission gas release data should be acceptable up to 85 GWd/MTU.

3.1.9 Radial Power Profile

As noted in Section 2.1, the radial power profile in the fuel pellet gets more edge-peaked with increasing burnup. This radial power profile is also impacted by the enrichment of the fuel. Most models for fuel radial power profile are derived from physics-based neutronics calculations and have been assessed against radial burnup and power profile measurements from the measurements of long-lived and short-lived fission products across the radius of a fuel pellet (Lassman, O'Carrol, VanderLaar, & Walker, 1994) (Lassman, Walker, & van de Laar, 1998). There is no concern that these models will cease to be valid at high burnup, but radial burnup and power measurements should be used to assess the accuracy of any radial power profile model.

A radial power profile model derived from physics-based neutronic calculations and verified using radial measurements of fission products that can be correlated to power and burnup should be acceptable up to 85 GWd/MTU.

3.1.10 Fuel Radial Relocation

Fuel radial relocation is a phenomenon that occurs early in life. During the first rise to power, or one of the subsequent rises to power, the fuel pellet experiences a very large temperature gradient radially

(~800°C) and the differences in thermal expansion causes cracking in the relatively brittle pellet (Galbraith, 1973) (Cunningham & Beyer, 1984). The cracked pellet has an effectively larger radius and thus the fuel/cladding gap is reduced. When the fuel/cladding gap closes, the pellet fragments have somewhat re-arranged during irradiation and although some of this radial relocation is recovered, all the relocation is not. Relocation is not a high burnup effect, except as far as there may be some evidence that at high burnup some of the gaps and cracks between the fuel fragments may begin to heal and the volume available for gas between the fragments in these areas may no longer be available. (Geelhood & Goodson, 2019) Not accounting for this could lead to an overestimation of rod void volume, which would lead to an underestimation of the rod internal pressure which would be non-conservative.

Current fuel radial relocation models should be acceptable up to 85 GWd/MTU. However, void volume data from high burnup fuel rods should be used to assess the ability of the code in question to predict the remaining void volume and subsequent pressure in high burnup rods.

3.1.11 High Burnup Rim Formation

As mentioned in Section 2.1, as burnup progresses beyond a rod-average burnup of 40 GWd/MTU, a so-called high burnup rim begins to form in the pellets that exhibits small ($< 1\mu\text{m}$) grains with low-angle grain boundaries and high porosity (Manzel & Walter, 2000) (Spino, Vennix, & Coquerelle, 1996) (Une, Mogita, Shiratori, & Hayashi, 2001). Although the explicit modeling of the high burnup rim thickness and characteristics is not strictly necessary in the performance of fuel safety reviews, the knowledge that such a rim exists is necessary in the identification of potential changes to fuel performance that could occur at high burnup such as fission gas release, solid and gaseous fuel swelling and pellet fragmentation following ballooning and burst.

3.1.12 Decay heat

Decay heat that is caused by fission products and actinides in fuel is a function of burnup and must be properly accounted for to perform both accident analyses and evaluations of spent fuel storage and transportation. The decay heat is typically calculated using a code such as ORIGEN and has been found to result in acceptably accurate predictions of decay heat. Current built-in libraries for ORIGEN go to a rod average burnup of 72 GWd/MTU. For application to higher burnup, it is recommended that new cross section sets be generated using a neutronics code such as SCALE or MCNP. We have no reason to believe the fidelity of these models would be reduced at high burnup up to 85 GWd/MTU relative to existing predictions up to 72 GWd/MTU.

Table 3.1. Tests that could be used to quantify property correlations for high burnup fuel

Property	Recommended Tests
Thermal conductivity	Tests on irradiated material recommended. Laser flash diffusivity on irradiated disks or pellets. Tests should be performed at irradiation temperature to avoid annealing
Thermal expansion	Not necessary. Tests on unirradiated samples acceptable
Emissivity	Not necessary
Enthalpy and specific heat (only needed for transient analysis and calculation of stored energy)	Not necessary. Tests on unirradiated samples acceptable
Melting Temperature	No further tests needed. Melting tests on irradiated samples already exist to high burnup.
Densification	Not necessary, BOL phenomenon
Swelling	No further tests needed. Could deduce from pellet swelling measurements. Significant data from high burnup pellets exist
Fission gas release	Puncture tests during PIE. Limited high burnup data exists. In-pile sweep tests to evaluate release of radioactive gas
Void volume	Puncture tests during PIE. Limited high burnup data exists.
Radial power profile	Tests on irradiated high burnup pellets would be useful. Radial Electron Probe Micro-Analysis (EPMA)
Fuel radial relocation	Not necessary, BOL phenomenon
High burnup rim formation	Not necessary. May be characterized by pellet microscopy. Does not directly impact safety, but could impact a number of other high burnup phenomena.

3.2 Cladding Material Property Correlations

The following cladding material properties are typically needed to perform fuel thermal-mechanical analysis of nuclear fuel with Zr-alloy cladding under normal conditions and AOOs:

- Thermal conductivity
- Thermal expansion
- Emissivity
- Enthalpy and specific heat
- Elastic modulus
- Yield stress
- Thermal and irradiation creep rate
- Axial irradiation growth
- Oxidation rate
- Hydrogen pickup.

The following additional material properties are typically needed to perform fuel-mechanical analysis of nuclear fuel under accident conditions based on the development and qualification of the NRC transient fuel performance code, FRAPTRAN (Geelhood K. , Luscher, Cuta, & Porter, 2016):

- High temperature ballooning behavior
- High temperature (800-1200°C) steam oxidation rate.

The cladding material properties used in FRAPCON are described in a materials property handbook (Luscher, Geelhood, & Porter, 2015) and have been applied up to a rod average burnup of 62 GWd/MTU. The burnup applicability to each property should be assessed separately. Table 3.2 provides a summary of the tests that could be performed to quantify the material properties discussed below.

3.2.1 Thermal Conductivity

Cladding thermal conductivity is not expected to change significantly with irradiation based on the currently available data. Typically heat transfer in a metal is due to electronic heat transfer (Gaskell, 1995), which is not significantly impacted by lattice damage done by fast neutron irradiation. No change in thermal conductivity with irradiation is used in FRAPCON (Luscher, Geelhood, & Porter, 2015). Thermal conductivity data as a function of temperature from unirradiated samples have typically been used to develop cladding thermal conductivity correlations.

A cladding thermal conductivity model based on unirradiated data should be acceptable up to 85 GWd/MTU.

3.2.2 Thermal Expansion

Cladding thermal expansion is not expected to change significantly with irradiation based on the currently available data. Thermal expansion is caused by crystal lattice expansion (Van Vlak, 1989) and does not change much with the introduction of dislocations from fast neutron irradiation. No change in thermal expansion with irradiation is used in FRAPCON (Luscher, Geelhood, & Porter, 2015). Thermal expansion data as a function of temperature from unirradiated samples have typically been used to develop cladding thermal expansion correlations.

A cladding thermal expansion model based on unirradiated data should be acceptable up to 85 GWd/MTU.

3.2.3 Emissivity

Cladding emissivity is important to calculate the portion of the gap heat transfer due to radiative heat transfer. The emissivity is impacted by the surface conditions, including any oxide on the surface of the cladding. Although oxide thickness increases with increasing burnup, the effect of oxide saturates at very low oxide thicknesses (Luscher, Geelhood, & Porter, 2015).

A cladding emissivity model based on existing lower-burnup data should be acceptable up to 85 GWd/MTU.

3.2.4 Enthalpy and Specific Heat

Cladding enthalpy and specific heat are not expected to change significantly with irradiation based on the currently available data. Specific heat of a material is primarily dependent on the composition and the crystal structure (Gaskell, 1995) and does not change much with the introduction of dislocations from fast neutron irradiation. No change in enthalpy or specific heat with irradiation is used in FRAPCON (Luscher, Geelhood, & Porter, 2015). Enthalpy and/or specific heat data as a function of temperature from unirradiated samples have typically been used to develop cladding enthalpy and specific heat correlations.

A cladding enthalpy and specific heat model based on unirradiated data should be acceptable up to 85 GWd/MTU.

3.2.5 Elastic Modulus

Cladding elastic modulus has been observed to be a weak function of fast neutron fluence (proportional to fuel burnup) (Geelhood, Beyer, & Luscher, 2008). Not all applicants include a fluence dependence in their safety analysis codes and methods. The impact of fluence on elastic modulus should be revisited and justified with data from cladding tubes that have been subjected to fast neutron fluence levels that are representative of high burnup.

A cladding elastic modulus model that either is or is not a function of fast fluence should be justified based on high burnup cladding data. A model based on irradiated cladding modulus data should be acceptable up to 85 GWd/MTU.

3.2.6 Yield Stress

Cladding yield stress has been observed to be a strong function of fast neutron fluence (proportional to fuel burnup) early in life and saturates to a value at moderate fluence levels (Geelhood, Beyer, & Luscher, 2008). It is expected that saturation will continue at high burnup. Temperature dependent data from irradiated tubes should be provided to justify saturation of the strength increase up to fast neutron fluence levels corresponding to burnup levels being requested.

A cladding yield stress model should be justified based on high burnup cladding data. A model based on irradiated cladding yield stress data should be acceptable up to 85 GWd/MTU.

3.2.7 Thermal and Irradiation Creep Rate

The creep behavior of zirconium-alloy tubes has often been characterized by a thermal rate that can be developed based on ex-reactor creep tests, which are a function of stress and temperature, and an irradiation rate (Matsuo, 1987) (Limbäck & Andersson, 1996), which can be developed based on the additional creep observed at the same stress and temperature during an in-reactor creep test. This creep rate can change significantly with small changes to alloy composition or microstructure. The increase or decrease in the thermal creep rate does not directly correlate to an increase or decrease in the irradiation creep rate. One example of this is the creep rates for recrystallized cladding and stress-relief annealed cladding in FRAPCON. Although both the thermal and irradiation creep rates are greater for the stress-relief annealed cladding than the recrystallized cladding, the two increases are not the same fraction so one increase could not be determined from the other (Geelhood K. , Luscher, Raynaud, & Porter, 2015) (Limbäck & Andersson, 1996). Both in-reactor and ex-reactor creep tests are recommended to justify the cladding creep correlation used, as these processes appear to be independent.

As irradiation progresses the radiation damage hardens the Zr-alloy cladding such that the thermal creep rate is often modeled as a function of fast neutron fluence (proportional to burnup). The irradiation creep rate is observed to be a function of fast neutron flux (proportional to linear heat generation rate). Cladding creep models that will be applied to high burnup cladding should be assessed against in-reactor creep data at representative fast neutron fluence levels.

A cladding thermal and irradiation creep rate model based on in-reactor creep data should be acceptable up to 85 GWd/MTU.

3.2.8 Axial Irradiation Growth

Zirconium-alloy tubes have been observed to grow axially with increased fast neutron fluence (Luscher, Geelhood, & Porter, 2015). This growth rate can change significantly with small changes to alloy composition or microstructure (for example, Zircaloy-2, Zircaloy-4, M5®, ZIRLO) (Irisa, et al., 2000) (Gilbon, Soniak, Doriot, & Mardon, 2000) (Amaya, Kakiuchi, & Mihara, 2019). Below 62 GWd/MTU, the axial irradiation growth rate appears to increase according to a power law. However, there is some indication that at higher burnup levels the irradiation growth rate may increase for some alloys. Cladding axial irradiation growth models that will be applied to high burnup cladding should be assessed against in-reactor growth data at representative fast neutron fluence levels.

A cladding axial irradiation growth model based on in-reactor growth data should be acceptable up to 85 GWd/MTU.

3.2.9 Oxidation Rate

The oxidation rate is important to model in cladding tubes as the zirconium oxide layer has a lower thermal conductivity than Zr metal. In addition, there is a safety limit that is applied to cladding regarding the maximum oxide layer thickness. Beyond this maximum thickness, the oxide has the potential to spall, creating local cool spots to which hydrogen preferentially diffuses. This can lead to the formation of a brittle hydride lens. Additionally, the oxide layer grows at the expense of the metallic cladding and the thinning of the cladding can lead to increased stress within in the cladding.

In the zirconium-alloy systems, ex-reactor autoclave corrosion data is significantly different from in-reactor corrosion data and should not be used to develop corrosion correlations for cladding tubes. Additionally, the corrosion behavior of non-fueled cladding segments may also not be representative of fueled cladding corrosion as the surface heat flux in the fueled cladding seems to strongly impact oxidation rate (Cox, 2005) (Sabol, Comstock, Weiner, Larouere, & Stanutz, 1993) (Garde, Pati, Krammen, Smith, & Endter, 1993).

The oxidation thickness increases with time and is typically modeled as a function of heat flux, cladding temperature, and alloy composition. After an initial formation of around 5 μm of oxide, the oxide build-up typically proceeds at a linear rate. This rate is different for different cladding alloys. At high burnup, the rate appears to deviate from a linear increase and begins to increase at an exponential rate. For Zircaloy-4, this occurs between 40 and 50 GWd/MTU. For ZIRLO™ this occurs around 50-60 GWd/MTU. Optimized ZIRLO™ and M5® appear to continue a linear increase beyond 70 GWd/MTU (Motta, Couet, & Comstock, 2015), but the transition to an exponential increase could occur above this level. Cladding oxidation models that will be applied to high burnup cladding should be assessed against in-reactor oxide thickness data from fuel rods that have been irradiated to high burnup. The high burnup data should consist of a significant range of temperatures and heat flux values.

A cladding oxidation model based on post-irradiation oxide thickness data should be acceptable up to 85 GWd/MTU.

3.2.10 Hydrogen Pickup

It is important to quantify the hydrogen pickup in cladding tubes as hydrides in zirconium can lead to brittle behavior of the cladding (Zhao, et al., 2017). There is a safety limit that is applied to cladding regarding the maximum hydrogen content. In PWR cladding, hydrogen is typically modeled with a constant pickup fraction (Geelhood & Beyer, 2011). In this case, when the oxidation rate goes up, as discussed previously, the hydrogen content will also go up. This may not be the case with increasing burnup and should be verified using PIE data on irradiated cladding. In BWR cladding, the hydrogen content is observed to increase at an exponential rate at high burnup. The expected hydrogen content should be verified from PIE data on irradiated cladding.

A cladding hydrogen pickup model based on post-irradiation oxide thickness and cladding hydrogen measurements should be acceptable up to 85 GWd/MTU.

3.2.11 High Temperature Ballooning Behavior

The burst stress as a function of temperature is important to know for LOCA analysis as this will determine when to start two-sided oxidation. The ballooning strain is important to determine flow blockage and establish if a coolable geometry has been maintained. Ex-reactor burst tests at temperatures of interest for LOCA on representative cladding segments have been used in the past to establish the high temperature ballooning behavior of Zr-alloy tubes (Hagrman, 1981) (Powers & Meyer, 1980). These tests are expected to be acceptable to establish ballooning behavior for burnup up to 85 GWd/MTU.

3.2.12 High Temperature Steam Oxidation Rate

The steam oxidation rate is important for LOCA analysis (in the range of 800-1200°C) because this determines if the cladding has been overly thinned by corrosion. This also determines the extra heat generation from the corrosion reaction (Baker & Just, 1962) (Cathcart, et al., 1977). The high temperature steam oxidation rate does not appear to be impacted by burnup. The use of autoclave data for this should be acceptable for use with cladding up to a burnup of 85 GWd/MTU.

Table 3.2. Tests that could be used to quantify property correlations for high burnup cladding

Property	Recommended Tests
Thermal conductivity	No new testing recommended because this property is not expected to be impacted by burnup. Tests on unirradiated cladding samples over representative temperature range are sufficient.
Thermal expansion	No new testing recommended because this property is not expected to be impacted by burnup. Tests on unirradiated cladding samples over representative temperature range are sufficient
Emissivity	No new testing recommended because this property is not expected to be impacted by burnup.
Enthalpy and specific heat (only needed for transient analysis and calculation of stored energy)	No new testing recommended because this property is not expected to be impacted by burnup. Tests on unirradiated cladding samples over representative temperature range are sufficient
Elastic Modulus	Tensile tests on irradiated cladding tubes over a representative range of burnup and temperatures is recommended to verify modest increase with fluence
Yield Stress	Tensile tests on irradiated cladding tubes over a representative range of burnup and temperatures is recommended to verify saturation of yield stress at high fluence.

Property	Recommended Tests
Thermal and irradiation creep	In-reactor creep tests on pressurized cladding tubes over a representative range of burnup and temperatures is recommended to verify expected behavior
Axial irradiation growth	Poolside length measurements from lead test assemblies (LTAs) over a representative range of burnup and temperatures is recommended. Specifically tests for possible high fluence increases.
Oxidation rate	Poolside eddy current measurements from LTAs over a representative range of burnup and temperatures to tests for increases at high burnup is recommended
Hydrogen pickup	Destructive examination of cladding segments from LTAs is recommended to verify hydrogen pickup models.
High temperature ballooning behavior	<p>No new testing recommended because this property is not expected to be impacted by burnup. Ex-reactor burst tests at relevant temperature. Unirradiated samples are acceptable.</p> <p>Ballooning behavior would possibly need to be better understood if it were to be used in a future assessment of fuel fragmentation, releases and dispersal</p>
High temperature steam oxidation rate	Ballooning behavior would possibly need to be better understood if it were to be used in FFRD. Ex-reactor corrosion tests at relevant temperature. Unirradiated samples are acceptable

3.3 SAFDL Limits for High Burnup Fuel

The previous section discussed the cladding material properties that are typically needed to perform the required thermal-mechanical safety analyses. The second step is determining the SAFDLs. The NRC Standard Review Plan Section 4.2 (US Nuclear Regulatory Commission, 2007) identifies several general phenomena that should be considered for standard fuel and cladding to avoid fuel system damage and fuel rod failure and to ensure fuel coolability. The standard review plan also provides some general guidance in selecting specific limits in each area. However, it is the responsibility of the applicant to propose and justify the specific limit that should be used in each area. It is also the responsibility of the applicant to identify and propose limits for possible damage mechanisms that have not been identified by the standard review plan.

To provide assistance to NRC staff during the review of an LTR or LTR supplement regarding high burnup, this section discusses the expected impact of burnup from 62 to 85 GWd/MTU on typical limits for Zr-alloy cladding and UO₂ fuel and the data that would likely need to be collected to justify a revised limit. This section also identifies new damage mechanisms that should be considered for high burnup fuel and cladding and identifies data collection that could be used to justify a new limit.

The SAFDLs mentioned in the standard review plan are broadly separated into three general categories:

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

Table 3.3 lists each of the SAFDLs mentioned in the standard review plan in each of these three general categories. Also shown in this table is the purpose of each established limit.

Table 3.3 SAFDLs from the standard review plan and the purpose of each limit

SAFDL Category	SAFDL	Purpose of Limit
Assembly Performance	Rod bow	Could impact DNBR or MCPR
	Irradiation growth	Excessive assembly growth could lead to assembly deformation
	Hydraulic lift loads	The weight of the assembly and force of holddown springs should prevent assembly liftoff
	Fuel assembly lateral deflection	Lateral deflections should not be so great as to prevent control rods/blades from being inserted or alter local thermal hydraulic conditions
	Fretting wear	Excessive fretting wear can lead to failed cladding
Fuel rod performance Normal operation and AOO	Cladding stress	Prevent failure of cladding from overstress conditions
	Cladding strain	Prevent failure of cladding from excessive strain conditions
	Cladding fatigue	Prevent failure of cladding from cyclic fatigue
	Cladding oxidation, hydriding and CRUD	Prevent oxide spallation which can result in formation of brittle hydride lens

SAFDL Category	SAFDL	Purpose of Limit
		Retain cladding ductility as stated in cladding strain limit
	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 departure of nucleate boiling (DNB)
	Internal hydriding	Retain cladding ductility as stated in cladding strain limit
	Cladding collapse	Prevent failure of cladding due to collapse in the plenum and axial pellet gaps which results in large local strains
	Overheating of fuel pellets	Prevent fuel melting to assure that axial or radial relocation of molten fuel would neither allow molten fuel to contact the cladding nor produce local hot spots. Fuel melting can cause large fuel expansion that will strain the cladding.
	Pellet-to-cladding interaction	Prevent cladding failure from chemically assisted cracking
Fuel rod performance (accident conditions)	Overheating of the cladding	Failure of cladding and dose consequence if critical heat flux is exceeded
	Excessive fuel enthalpy	Failure of cladding and dose consequence during RIA if injected energy limit is exceeded
	Bursting	Time of burst during LOCA needed for oxidation of inner cladding and associated heat is correctly modeled
	Mechanical fracturing	Failure of cladding and dose consequence from external event
	Cladding embrittlement	Coolable geometry must be retained following LOCA and non-LOCA events
	Violent expulsion of fuel	Coolable geometry must be retained following RIA Pressure pulse must not damage reactor vessel
	Generalized cladding melting	Coolable geometry must be retained following LOCA and non-LOCA events

SAFDL Category	SAFDL	Purpose of Limit
	Fuel rod ballooning	Degree of ballooning needed to calculate blockage of the coolant channel
	Structural deformation	Coolable geometry must be retained following LOCA or seismic event

The existing SAFDLs and any additional concerns are described in the following sections, which are devoted to each one of the three general categories above. Table 3.4 provides a summary of the tests that could be performed to justify the SAFDLs discussed below beyond what was necessary to quantify the material property correlations.

3.3.1 SAFDLs Related to Assembly Performance

SAFDLs related to assembly performance are typically performed by simple hand calculations or by citing manufacturing controls or historic data. The only assembly SAFDL that is not addressed in this way, is lateral distortion to external loads. This analysis requires testing, model development, and detailed calculations. The limits that are typically used for fuel up to 62 GWd/MTU are likely appropriate, but some of the empirical methods likely need more data to justify their use to 85 GWd/MTU.

3.3.1.1 Rod Bow

Usually there is a penalty on departure from nucleate boiling ratio (DNBR) or margin to critical power ratio (MCPR) to account for bowing. The limits of what degree of bowing is acceptable will not change with increasing burnup as this is controlled by the physical dimensions of the fuel assembly. However, bowing methods rely on correlations that are very empirical. Some testing or assessment would be useful to assess the applicability of the rod bow correlation used for high burnup fuel.

3.3.1.2 Irradiation Growth

The assembly design allows for a given amount of growth and will define the limit. The axial growth correlation from Section 3.2.8 will be used to assess maximum growth. There are currently no additional concerns that need to be addressed regarding irradiation growth for high burnup fuel.

3.3.1.3 Hydraulic Lift Loads

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. Typically, the holddown springs exhibit relaxation with increasing burnup. This is accounted for in current methods, but the spring relaxation should be verified at high burnups.

3.3.1.4 Fuel Assembly Lateral Deflections

The limits for fuel assembly lateral deflections are such that the control rod (PWR) or control blades (BWR) can still be inserted as needed. Current assembly and channel bow methods are used to assess performance relative to these limits. Assembly and channel bow are not impacted by fuel rod performance, but rather by channel design (BWR) and guide tube design (PWR) and therefore these limits and methods should be assessed against data from high burnup assemblies. Higher burnups translate to more fluence-induced guide tube growth and fluence-induced channel distortion. Longer reload cycles may also mean more control blade insertion (to hold down excess reactivity) which could result in more channel distortion due to shadow corrosion.

3.3.1.5 Fretting Wear

Current design limits state that fuel rod failures will not occur due to fretting. Fretting has historically been controlled through debris filters that reduce the possibility for debris fretting and through spacer design to reduce fretting between fuel rods and grid features. Surveillance of high burnup fuel assemblies should be performed to demonstrate that fretting failure will continue to be mitigated by these strategies at high burnup.

3.3.2 SAFDLs Related to Rod Performance Assessed for Normal Operation and AOOs

Current codes that are informed by the properties in Section 3.1 and 3.2 can perform the following analyses. However, for application to 85 GWd/MTU, the limits may need revision relative to those typically used for fuel up to 62 GWd/MTU. Several of these SAFDLs also have application in accident analysis.

3.3.2.1 Cladding Stress

Cladding stress limits are typically set using a method described in Section III of the ASME code (American Society of Mechanical Engineers, 2017). Typically, these limits are based on unirradiated yield stress to represent the lowest yield stress. Even in the case that these limits are based on irradiated yield stress, the increase in yield stress with fast neutron fluence saturates, such that further increase in yield stress at high burnup is not expected. See Section 3.2.6. A new cladding stress limit is not expected for high burnup fuel.

3.3.2.2 Cladding Strain

There are two cladding strain limits that are typically employed. The first steady-state limit is the maximum positive and negative deviation from the unirradiated conditions that the cladding may deform throughout life. The second transient strain limit is the maximum strain increment caused by a transient. These cladding strain limits are typically justified based on mechanical tests (axial tension tests and tube burst tests) performed on irradiated cladding tubes. Ductility tends to decrease with irradiation (Geelhood, Beyer, & Luscher, 2008), so these tests are most relevant when performed at the maximum expected fast neutron fluence. In the case of operation up to 85 GWd/MTU, this maximum fluence will be extended from what it had been. The uniform elongation or strain away from the rupture has been

typically used as the strain capability for Zr-based alloys (Geelhood, Beyer, & Cunningham, 2004). Typical cladding strain limits that had been applied to fuel up to 62 GWd/MTU should be re-evaluated against ductility data from high burnup cladding tubes to see if these limits are still representative of the cladding strain capability of high burnup cladding. Certain cladding alloys may exhibit more or less ductility at high burnup, so data from the cladding alloy in question should be used to justify the high burnup strain limits. It would also be acceptable to propose a lower strain limit to be used when fuel exceeds a certain burnup level.

3.3.2.3 Cladding Fatigue

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. The damage fractions are typically found relative to the O'Donnell and Langer irradiated fatigue design curve (O'Donnell & Langer, 1964). During past burnup extensions, there was no appreciable drop in fatigue life with increasing burnup/fast neutron fluence. Some data from high burnup cladding such as (Soniak, Lansart, Royer, Mardon, & Waeckel, 1994) (Wisner, Reynolds, & Adamson, 1994) should be used to confirm the O'Donnell and Langer irradiated fatigue design curve up to 85 GWd/MTU.

3.3.2.4 Cladding Oxidation, Hydriding, and CRUD

For Zr-alloy cladding, the cladding oxidation limit is designed to preclude oxide spallation that has typically been observed above 100 μm . Oxide spallation can lead to a local cool spot which acts as a sink for hydrides, creating a local, extremely brittle hydride lens (Motta, Couet, & Comstock, 2015). The hydrogen limit is designed to ensure that the strain limit previously identified will be applicable since high levels of hydrogen (>600ppm) can cause embrittlement of the cladding (Geelhood, Beyer, & Luscher, 2008). There is no explicit limit on CRUD¹, other than it be explicitly considered if it is present and it is typically modeled as an insulating layer around the fuel rod in plants that have CRUD issues.

For application up to 85 GWd/MTU, these existing limits are acceptable, but there is an increasing chance that more rods will approach these limits. Therefore, the NRC reviewers should apply increased scrutiny to the codes and methods used to assess the cladding oxidation, hydriding, and CRUD for safety analyses.

3.3.2.5 Fuel Rod Internal Pressure

There are several possible limits for rod internal pressure that are discussed in the Standard Review Plan Section 4.2 (US Nuclear Regulatory Commission, 2007). The first and most straightforward is that the rod internal pressure shall not exceed the coolant system pressure. No outward deformation or hydride reorientation is possible if the stress in the cladding is in the compressive directions. This situation does not change with increased burnup levels.

Greater rod internal pressures may be justified based on the following criteria:

- No cladding liftoff during normal operation
- No reorientation of the hydrides in the radial direction in the cladding

¹ Chalk River Unknown Deposit (generic term for deposits on fuel cladding)

- A description of any additional failures resulting from departure of nucleate boiling (DNB) caused by fuel rod overpressure during transients and postulated accidents.

There may also be a relationship between limits on rod internal pressure and assumptions related to iodine scrubbing (decontamination) factors in the fuel handling accident dose assessment.

It has typically been determined by applicants with Zr-alloy cladding that the first of these criteria, no cladding liftoff during normal operation, is the most limiting. This should be confirmed by the applicant of a burnup extension to still be the case. If this is found to be the case, the pressure limit where cladding liftoff could occur is typically set as the pressure where the upper bound cladding creep rate will exceed the lower bound fuel pellet swelling rate. For high burnup fuel, the fuel pellet swelling rate will be determined as discussed in Section 3.1.7 and the cladding creep rate will be determined as discussed in Section 3.2.7. For application up to 85 GWd/MTU, these existing limits are acceptable, but there is an increasing chance that more rods will approach these limits. Therefore, the NRC reviewers should apply increased scrutiny to the codes and methods used to assess the fuel rod internal pressure for safety analyses.

3.3.2.6 Internal Hydriding

Internal hydriding is typically addressed through manufacturing controls on the pellet moisture limit. This limit and the way it is addressed is not expected to change for burnup up to 85 GWd/MTU.

3.3.2.7 Cladding Collapse

Cladding collapse in modern nuclear fuel rods 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. Nevertheless, cladding collapse analyses are performed for potential small axial gaps between pellets and in the upper plenum region. The key input into this analysis is the cladding creep rate. The worst-case situation for cladding collapse is beginning of life when the internal pressure is the lowest. Therefore, this limit and its analysis will likely not be affected for operation up to 85 GWd/MTU.

3.3.2.8 Overheating of Fuel Pellets

For this analysis, the limit is the melting temperature of the fuel pellets. As discussed in Section 3.1.5, the melting temperature is expected to decrease with increasing burnup (Popov, Carbajo, Ivanov, & Yoder, 2000). Additionally, the fuel thermal-mechanical code used to assess the power to melt should be assessed to give adequate fuel temperatures with adequate uncertainty levels up to 85 GWd/MTU. The existing fuel melting limit that has a decreasing melting temperature with burnup is likely to stay the same but additional scrutiny should be applied to the codes, methods, and uncertainties used to calculate the fuel temperature at high burnup.

3.3.2.9 Pellet-to-Cladding Interaction

Typically, there is 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. It is expected

that this approach will be acceptable to 85GWd/MTU, but a surveillance plan should be enacted to ensure that rods are not failing by a mechanism driven by pellet-to-cladding interaction.

3.3.3 SAFDLs Related to Fuel Rod Performance Assessed for Accident Conditions

Current codes that are informed by the properties in Section 3.2 can perform the following analyses. However, the limits may need revision for application to high burnup.

Work is currently underway to change some regulations (10CFR50.46(c)) and staff guidance (DG-1327) for design basis event analysis including LOCA and RIA. Neither of these are complete yet, so the discussion in this report will reflect the current regulations and staff guidance.

3.3.3.1 Overheating of the Cladding

Overheating of the cladding refers to exceeding the critical heat flux (CHF). Operation above this point results in a reduction of the ability of the coolant to remove heat and can result in damage to the cladding. In a PWR, exceeding CHF results in DNB. In a BWR, exceeding CHF results in dryout. This thermal margin should not be exceeded for normal operation and AOOs. For design basis accidents the number of fuel rods exceeding thermal margin criteria are assumed to have failed and are included in fission product release dose calculations.

The boiling transitions are shown graphically in Figure 3.1. Typical limits are based on ex-reactor flow tests on electrically heated fuel assembly mockups to determine where CHF occurs. The CHF is primarily influenced by the geometry of the assembly, although surface conditions of the fuel rods may also impact the CHF. These correlations are currently assumed to be applicable at all burnup levels. There is no indication that this would not continue to be the case between 62 and 85 GWd/MTU.

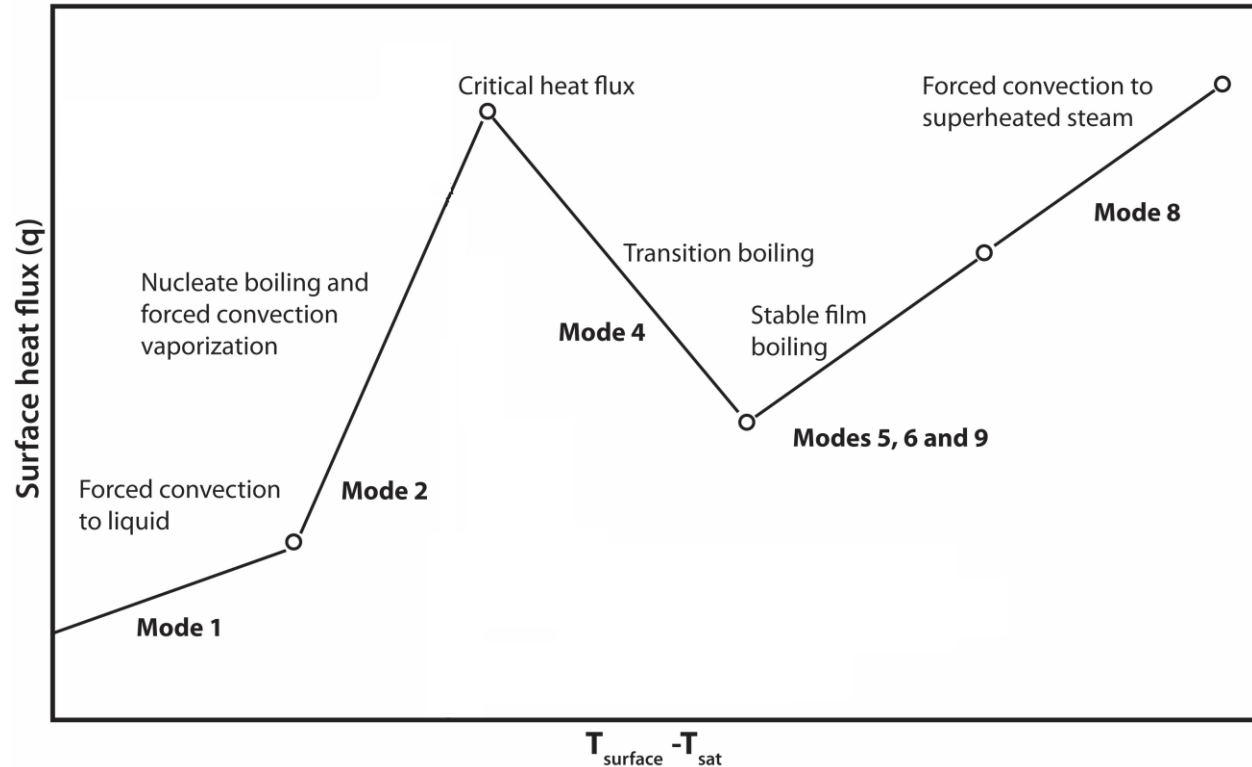


Figure 3.1. Typical boiling transitions

3.3.3.2 Excessive Fuel Enthalpy

Excessive fuel enthalpy relates to the sudden increase in fuel enthalpy from an RIA below the fuel melting limit that can result in cladding failure due to pellet-cladding mechanical interaction. NRC Draft Guide 1327 (US Nuclear Regulatory Commission, 2016) discusses other failure modes driven by excessive fuel enthalpy such as oxidation induced embrittlement, and ballooning and burst. 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 has been determined of what level of fuel enthalpy increase will cause cladding failure (Mihara T. , Udagawa, Amaya, & Kakiuchi, 2019). Data regarding cladding failure during RIA has been collected up to a burnup level of 80 GWd/MTU (Geelhood & Luscher, 2016). However, it has been observed to be impacted by alloy composition and heat treatment and cladding hydrogen concentration.

An alternate approach comes from the fact that cladding failure due to excessive fuel enthalpy is driven by pellet-cladding mechanical interaction which causes the cladding to exceed its ductility limit. Therefore, it is possible to collect uniform elongation (strain at maximum load) data from the irradiated cladding mechanical tests. If it can be shown that the high burnup cladding in question has a beneficial or negligible impact on the uniform elongation relative to the alloy that has been tested, then it could be reasonably argued that the current RIA failure limits are applicable to the high burnup cladding in question. However, it is noted that as burnup increases, the hydrogen content will increase and this will drive the cladding ductility down. (Geelhood, Beyer, & Luscher, 2008). Large oxide layers will thin the cladding, effectively reducing the thickness of the ductile cladding.

It should be noted that this limit is used to assess the number of fuel rods that are expected to fail during an RIA, and a conservative approach could be taken to either assume all the rods will fail or a significantly conservative limit could be applied to cover the lack of alloy-specific high burnup RIA test data.

3.3.3.3 Bursting

Bursting of the fuel rod relates to failure of fuel rods due to high temperature and high gas pressures during a LOCA. 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 not be underestimated and the balloon 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.

An applicant will typically use an empirical correlation for burst stress and ballooning strain such as the one given in NUREG-0630 (Powers & Meyer, 1980). Typically, unirradiated data is used to develop burst correlations. At the ballooning and burst temperature, most of the radiation damage is expected to anneal out (Hagman, Reymann, & Mason, 1981). This is expected to remain applicable for high burnup cladding between 62 and 85 GWd/MTU.

3.3.3.4 Mechanical Fracturing

Mechanical fracturing refers to a defect in the cladding caused by an externally applied force. Typically, this limit has conservatively been set as applied stresses above 90% of the irradiated yield stress. This limit should not be exceeded for normal operation and AOOs. For design basis accidents 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 high burnup fuel between 62 and 85 GWd/MTU given that the irradiated yield stress obtained as described in Section 3.2.6 is used.

3.3.3.5 Cladding Embrittlement

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. 10 CFR 50.46 specifies a cladding temperature limit of 2200°F (1204°C) and a peak oxidation of 17% equivalent cladding reacted for Zr-alloy cladding (US Nuclear Regulatory Commission, 2017).

There are several embrittlement mechanisms that manifest themselves at high burnup, chief among these, is hydrogen content in the cladding. Currently there is work being done at the NRC to change these regulations to preclude cladding embrittlement. Until this time, the applicant should propose one or more embrittlement criteria for high burnup fuel (62-85 GWd/MTU). Tests showing ductility at either these existing limits or test establishing new limits would be useful to demonstrate embrittlement will not occur. NRC has provided draft guidance on testing protocol for cladding embrittlement measurements in

Draft Guide 1262 (US Nuclear Regulatory Commission, 2019) and the development of analytical limits in Draft Guide 1263 (US Nuclear Regulatory Commission, 2019). In addition to the tests performed to establish the ballooning (Section 3.2.11) and high temperature oxidation behavior (Section 3.2.12), some prototypic integral LOCA tests (see for example (Flanagan, Askeljung, & Puranen, 2013)) where cladding tubes are subject to ballooning and burst in steam under expected time frames and samples are then subjected to mechanical loading such as bend tests after ballooning, burst, and high temperature oxidation are very useful to establish cladding embrittlement limits. For these tests, irradiated cladding tubes are preferable.

3.3.3.6 Violent Expulsion of Fuel

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. Typical limits for violent expulsion of fuel are:

- Peak radial average fuel enthalpy below 230 cal/g
- Peak fuel temperature below melting temperature.

It is expected that cladding failure will occur well before 230 cal/g for Zr-alloy. These limits are derived to prevent violent ejection of fuel from failed cladding. Given the propensity for fuel expulsion during high burnup LOCA tests, it is recommended that RIA tests also be examined to determine if fuel expulsion would occur at high burnup.

3.3.3.7 Generalized Cladding Melting

Generalized cladding melting is applicable to design basis accidents 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 Zr-alloy tubes the embrittlement limit of 1204°C (Section 3.3.3.5) is more limiting. The melting temperature of Zr-alloy cladding is not significantly impacted by radiation to high burnup levels.

3.3.3.8 Fuel Rod Ballooning

Ballooning of the fuel rod relates to failure of fuel rods due to high temperature and high gas pressures during a LOCA. 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 the degree of swelling not be underestimated and the balloon not block the coolant channel.

An applicant will typically use an empirical correlation for burst stress and ballooning strain such as the one given in NUREG-0630 (Powers & Meyer, 1980). Typically, unirradiated data is used to develop ballooning correlations (Hagrman, Zircaloy Cladding Shape at Failure (BALON2). EGG-CDAP-5379, 1981). This is expected to remain applicable for high burnup cladding between 62 and 85 GWd/MTU.

3.3.3.9 Structural Deformation

Structural deformation refers to externally applied loads during LOCA or an earthquake-triggered safe shutdown 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 design basis accidents 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 high burnup fuel given that the irradiated yield stress obtained as described in Section 3.2.6 is used.

3.3.4 New Damage Mechanism

The only new phenomenon that is observed at burnup between 62 and 85 GWd/MTU is the pellet fragmentation and dispersal that is observed during LOCA testing over this burnup range (Flanagan, Askeljung, & Puranen, 2013) (Oberlander & Wiesenack, 2014) (Raynaud P. , 2012). The consequence of large-scale fuel dispersal during a LOCA is that the coolable geometry of the fuel rods may be lost.

It is recommended that a new SAFDL be established by fuel vendors for the analysis of high burnup fuel (62-85 GWd/MTU). This limit should define an acceptable level of fuel dispersal to avoid a loss of coolable geometry and an analysis method should be developed regarding how to ensure that this limit will be met. This SAFDL should be applied to all design basis events. An alternate approach could be used to show that high burnup fuel does not fail during design basis events. Such an approach has been demonstrated for LOCA in (Hemlin & Nylen, 2019) (Pimentel & Smith, 2019).

Of particular concern with fuel fragmentation, is that failure of the fuel pellet can lead to a significant gas release that may not be otherwise accounted for that could affect the release assumptions for failed fuel.

Table 3.4. Tests that could be used to establish SAFDL limits on high burnup fuel beyond those needed to quantify basic material properties

SAFDL Limit	Recommended Tests
Rod Bow Evaluation	Ensure empirical rod bow method appropriately models LTA rods
Fretting	Ex-reactor tests on unirradiated tubes and grids to demonstrate no damage to either part
Cladding strain/ductility	Ex-reactor tests on irradiated tubes to confirm ductility requirements in strain limits at AOO temperatures
Cladding fatigue	Ex-reactor tests on irradiated tubes to establish fatigue design curve

SAFDL Limit	Recommended Tests
Excessive fuel enthalpy	Ex-reactor tests on irradiated tubes to confirm ductility requirements at RIA temperatures RIA tests on irradiated fuel segments in test reactor could be used to develop RIA failure criteria
Cladding Embrittlement	Ex-reactor balloon/burst/bend tests on irradiated fuel segments to confirm existing embrittlement limits or develop new embrittlement limits
Fuel Dispersal (New SAFDL)	Ex-reactor balloon/burst/bend tests on irradiated fuel segments to confirm quantity of fuel expelled. Criticality assessment to verify acceptability of this level of fuel expulsion

3.4 Changes to Existing Codes and Methodologies

This section discusses the changes to existing codes and methodologies.

3.4.1 Codes

As mentioned at the beginning of Section 3.0, in order to perform the safety analysis for fuel in a burnup range beyond where approval has previously been given, an applicant must submit a new or revised fuel performance code and safety analysis methodology. The material properties discussed in Sections 3.1 and 3.2 should be incorporated into a new or revised code. Following this, the critical need for the updated codes is validation. Validation of a computer code is typically performed on five areas that directly relate to various SAFDLs. These are also the areas used to assess FRAPCON (Geelhood & Luscher, 2015):

- Fuel temperature
- Fission gas release
- Rod internal pressure and void volume
- Cladding oxide thickness
- Cladding permanent hoop strain following a power ramp.

Details for each of these assessments is discussed in the following sections as they relate to the assessment of a code to correctly model high burnup fuel (62-85 GWd/MTU). Table 3.5 provides test data that could be used in code assessment.

3.4.1.1 Fuel Temperature

A fuel thermal-mechanical code will be used to assess the power to melt limit as well as provide initial conditions to accident analyses. Temperature data was historically collected in the Halden reactor in

Norway, which has recently been permanently shut down. Other reactor capabilities are being examined to determine how this capability can be replaced. There is currently some assessment data of temperature to high burnup (see Section 5.1.1). All available temperature data should be used to assess a fuel code's predictions of fuel temperature to the burnup level requested.

3.4.1.2 Fission Gas Release

Fission gas release is primarily driven by fuel temperature, time, and power level. Fission gas release can drive fuel temperatures and rod internal pressure, and as such is a key metric of success in a fuel thermal-mechanical code. There is currently some fission gas release data from high burnup fuel (see Section 5.2.1). Any fission gas release from destructive examination of high burnup lead test assemblies (LTAs), particularly high-power LTAs, would be useful in the assessment of a thermal-mechanical code used for safety analysis of high burnup fuel. An ongoing surveillance plan with the goal of continuing to obtain more fission gas release data would provide additional assessment data.

3.4.1.3 Rod Internal Pressure and Void Volume

A fuel thermal-mechanical code will be used to assess the rod internal pressure relative to the pressure limit that has been derived by the applicant. Void volume is impacted by component temperatures and deformations. The rod internal pressure is driven primarily by the void volume, fission gas release and component temperature and therefore could also be impacted.

Any void volume and rod internal pressure from destructive examination of high burnup LTAs, particularly high-power LTAs, would be useful in the assessment of a thermal-mechanical code used for safety analysis of high burnup cladding. An ongoing surveillance plan with the goal of continuing to obtain more void volume and rod internal pressure data would provide additional assessment data.

3.4.1.4 Cladding Oxide Thickness

Cladding oxide thickness is important because it can have a feedback on the fuel and cladding temperature predictions. Additionally, a fuel performance code will be used to assess the cladding oxide thickness relative to limits derived by the applicant. The data used to develop the cladding oxidation rate discussed in Section 3.2.9 is the same data that would be used to assess the code's prediction of oxidation rate. There is no additional requirement on validation beyond this. However, it should be noted that since high burnup operation would drive most cladding alloys closer to their limits, a surveillance plan to monitor the oxide thickness on high burnup fuel cladding would be useful as it may give an early indication of a burnup phenomenon that is causing a problem.

3.4.1.5 Cladding Permanent Hoop Strain Following a Power Ramp

A fuel thermal-mechanical code will be used to assess the cladding permanent hoop strain during an AOO power ramp to compare to the cladding strain limit. As burnup increases, new mechanisms such as gaseous swelling appear to have an impact on cladding strain following a power ramp. Additionally, increased cladding embrittlement may cause the cladding to fail prior to previously established strain limits. Therefore, power ramp tests on rodlets refabricated from irradiated high burnup fuel rods would

be helpful to assess the code prediction of hoop strain following a power ramp and establish high burnup cladding strain limits.

Table 3.5 Assessment data that could be used to validate fuel thermal-mechanical codes for high burnup (62-85 GWd/MTU) fuel

Assessment Data	Recommended Tests
Fuel centerline temperature	Use existing Halden data as well as any other data that can be obtained
Fission gas release	Existing data, LTA data, and follow-up surveillance plan
Rod internal pressure and void volume	Existing data, LTA data, and follow-up surveillance plan
Cladding oxide thickness	Initially none beyond data in Table 3.2. and follow-up surveillance plan
Cladding permanent hoop strain	Power ramp tests to assess the prediction of cladding strain following power ramp and high burnup cladding strain limits

3.4.2 Methodologies

The methodology for performing the fuel system safety analysis consists of the following pieces:

- Identification of functional requirements for the fuel and assembly
- Identification of limits for each functional requirement
- Identification of code or other approach that will be used to assess performance against functional requirement
- Identification of approach to demonstrate high level of confidence that design will not exceed functional requirements:
 - Selection of power histories to be considered
 - Identification of uncertainties in operational parameters
 - Identification of fabrication uncertainties
 - Identification of modeling uncertainties
 - Approach to quantify an upper tolerance level based on identified uncertainties.

The identification of SAFDLs for the fuel and assembly and the limits for each are discussed in Section 3.3. One new damage mechanism has been identified in Section 3.3.4 that should be implicitly handled via existing SAFDLs and considered in the development of those SAFDL limits. Alternatively, the

methodology may be modified to explicitly address these mechanisms through new functional requirements and limits.

The material property updates and the code assessment has been discussed in Sections 3.2 and 3.4.1. No further methodology change is anticipated as far as the use of codes is considered.

The identification of operational parameters such as rod power and coolant flow rate are not expected to be impacted by operation to higher burnup.

The identification of fabrication uncertainties will be taken from uncertainty specifications on the drawings or from manufacturing data. Although specific values may change, the general approach for obtaining these values is not expected to change.

The identification of modeling uncertainties should be developed and possibly increased for high burnup fuel to account for the relatively smaller data sets available to assess high burnup fuel performance. Comparing property data to correlations and code predictions to measurements should allow for the appropriate development of acceptable modeling uncertainties. These uncertainties should be revised or justified for high burnup.

Existing approaches to calculate upper tolerance levels are robust and should be acceptable to perform these calculations for high burnup fuel given that the activities discussed above are rigorously performed.

4.0 Storage and Transportation of Spent Nuclear Fuel

The safety analyses for storage and transportation of fresh and SNF is somewhat different than the in-reactor performance. The storage and transportation has not been historically considered as a part of the fuel design process. The in-reactor safety analysis which is part of the core design process is performed prior to each cycle. During this process, if an analysis indicated that any of the thermal-mechanical SAFDLs are exceeded a core redesign will be triggered. This process will continue until an adequate core loading pattern and operating plan has been determined. There are currently no in-reactor operating restrictions that are in place because of SNF considerations. Additionally, the individual rod power histories are not considered in the analyses of SNF during drying, loading, storage, or transportation. Therefore, the safety limits for storage and transportation of SNF must be developed for the most limiting fuel rods at the maximum expected burnup.

For storage and transportation of fresh fuel, there is currently nothing fundamentally different about fuel destined for high burnup operation than fuel destined for low burnup. At some point in the future, in order to routinely achieve high burnup, it is likely that the fuel enrichment will have to be increased beyond 5% ^{235}U . If this were to occur, this increase would have an impact on the manufacture, transportation, and storage of fresh nuclear fuel as current criticality analyses have been performed for fuel at 5% ^{235}U . However, the scope of this current report is to assess only the impact of high burnup on fuel storage and transportation and therefore there is no impact on the manufacture, transportation, and storage for fresh nuclear fuel if the enrichment of the fuel remains below 5% ^{235}U .

For spent fuel storage and transportation, burnup has a profound impact on the safety analyses. As burnup progresses, the cladding strength goes up and ductility goes down. Additionally, the cladding is effectively thinned by the corrosion process. This also leads to an increased cladding hydrogen concentration which embrittles the cladding. Additionally, the increased fission gas release leads to increased rod internal pressure and increased radioactive source term for accident analyses.

The following sections will provide a critical evaluation of the current limits that have been developed by the NRC for storage and transportation of SNF as they would apply to higher burnup. Previous burnup extension for burnup from 45 GWd/MTU to 62 GWd/MTU in 1999 will be used as a guide for these evaluations.

4.1 Wet Storage of Spent Nuclear Fuel

To meet the requirements of GDC 61, 2, 4, and 63, it is important that the spent fuel storage pool structures, systems, and components be designed to accomplish the following:

- prevent loss of water from the fuel pool that would lead to water levels that are inadequate for cooling or shielding
- protect the fuel from mechanical damage
- provide the capability to limit potential offsite exposures in the event of a significant release of radioactivity from the fuel or significant leakage of pool coolant
- provide adequate cooling to the spent fuel to remove residual heat

In the context of high burnup fuel, Regulatory Guide 1.13 (US Nuclear Regulatory Commission, 2007) identifies that the mechanical properties of the fuel and cladding may change with higher burnup. For

instance, high-burnup fuel may become more brittle (i.e., possess lower ductility and fracture toughness) and, therefore, be more vulnerable to failure. In order to protect high-burnup fuel from mechanical damage, this potential vulnerability should be considered in the design of spent fuel handling and storage facilities.

Additionally, the decay heat of high burnup SNF is likely greater and needs to be considered. As discussed in Section 3.1.12, we have no reason to believe the fidelity of the models used to calculate decay heat would be reduced at high burnup. It is noted, however, that longer wet storage time may be necessary for high burnup fuel assemblies prior to transfer to dry storage.

4.2 Dry Storage of Spent Nuclear Fuel

4.2.1 Current Regulatory Framework

The regulations in 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste,” (US Nuclear Regulatory Commission, 2015) include a number of fuel-specific and dry storage system (DSS)-specific requirements that may be dependent of the design basis condition of the fuel cladding. 10 CFR 72.44(c) states that a specific license for dry storage of SNF is to include technical specifications that, among other things, define limits on the fuel and allowable geometric arrangements. Additionally, 10 CFR 72.236(a) states that, a Certificate of Compliance (CoC) for a DSS design must include specifications for:

- the type of spent fuel (i.e., BWR, PWR, or both)
- maximum allowable enrichment of the fuel prior to any irradiation
- burnup
- minimum acceptable cooling time of the spent fuel before storage in the spent fuel storage cask
- maximum heat designed to be dissipated
- maximum spent fuel loading limit
- condition of the spent fuel (i.e., intact assembly or consolidated fuel rods)
- inerting atmosphere requirements.

The condition of the SNF cladding is critical to the storage as 10 CFR 72.122(h)(1) states that the SNF cladding is to be protected against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. Additionally, CFR 72.122(l) states that the DSS must be designed to allow ready retrieval of the SNF¹.

In addition to these regulations, the NRC staff have provided NUREG-1536, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General Licensing Facility” (US Nuclear Regulatory Commission, 2010) and NUREG-1567 “Standard Review Plan for Spent Fuel Dry Storage Facilities” (US Nuclear Regulatory Commission, 2000) . These standard review plans provides guidance in the preparation by the applicant and review by the staff of a topical report describing storage of SNF in a DSS and site-specific licenses. The standard review plan lays out the following evaluations that should be performed during the

¹ “Ready retrieval” is defined as an ability to retrieve a fuel assembly intact using its normal lifting hardware

safety evaluation of a DSS shown in Figure 4.1. Those areas impacted by high burnup are discussed in Section 4.2.2.

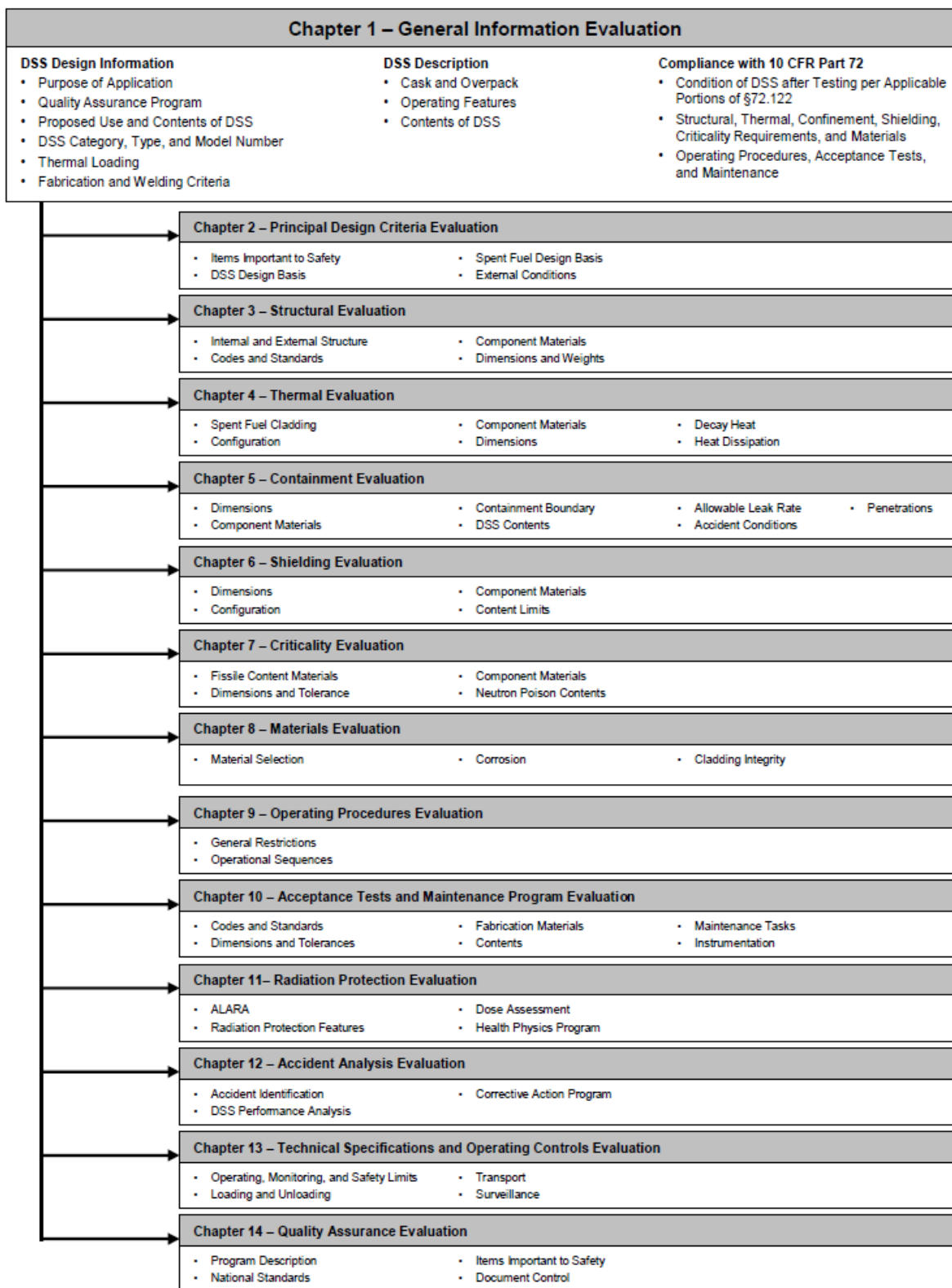


Figure 4.1. Overview of Safety Evaluation of a DSS (Taken from NUREG-1536 rev.1)

As burnups have progressed and new data have become available, NRC staff published Interim Staff Guidance 11 (ISG-11) regarding cladding considerations for the transportation and storage of spent fuel. This information is used to supplement the guidance in the standard review plan. The standard review plan (NUREG-1536 Rev. 1 (US Nuclear Regulatory Commission, 2010)) was revised in 2010 to reflect the latest guidance in ISG-11 Rev. 3. NUREG-1567 (US Nuclear Regulatory Commission, 2000) has not been revised to incorporate ISG-11 Rev. 3

ISG-11 has been periodically revised as more data have become available. The following summary lists the main guidance provided in each revision to ISG-11:

ISG-11 Rev.0

- Supplement the standard review plan by addressing potential degradation of high burnup fuel (> 45 GWd/MTU)

ISG-11 Rev. 1

- Incorporate new data
- Give applicant responsibility for demonstrating that the cladding is adequately protected
- Cladding oxidation should not be credited as load-bearing in the fuel cladding structural evaluation
- Defined a 1-percent creep strain limit on the cladding
- Only accounted for Zircaloy-clad fuel rods (no advanced cladding)

ISG-11 Rev. 2

- Changed the definition of damaged fuel
- Removed the 1-percent creep strain limit
- Discuss criteria to limit hydride reorientation in the cladding
- Applicable to all zirconium-alloy claddings and all burnup levels
- Described calculations to determine the maximum cladding temperature per a justified creep strain limit

ISG-11 Rev. 3

- Replaced calculation of maximum cladding temperature per a justified creep strain limit with a generic 400°C peak cladding temperature limit for NCS and NCT
- Allowed a higher short-term temperature limit for low-burnup fuel if it could be demonstrated that cladding hoop strain does not exceed 90 MPa
- Generic maximum cladding temperature limit of 570°C for off-normal and accident conditions applicable to all burnups
- Minimize hydride reorientation by restricting change in cladding temperature during drying operations to 65°C and the cladding should not experience more than ten thermal cycles each not exceeding 65°C.

NRC staff has recently provided a draft NUREG that is out for comment (NUREG-2224 (Ahn, et al., 2018)) regarding dry storage and transportation of high burnup SNF. This document seems to consider fuel with burnup up to 65 GWd/MTU. Additionally, NRC staff recently presented their thoughts on the management of high burnup spent fuel (Torres, 2018). With regard to storage of SNF, these documents discussed the roles of cladding creep and hydride reorientation. These documents also provided guidance for dry storage for less than 20 years and for dry storage for greater than 20 years.

Regarding operability and safety significance of aging mechanisms of the fuel cladding/assembly hardware on the performance of the fuel for dry storage periods up to 60 years, NRC has issued NUREG-2214 (US Nuclear Regulatory Commission, 2019) that provides the technical basis for these issues.

4.2.2 Application to Burnup Between 62 and 85 GWd/MTU

In examining the safety evaluations that should be performed on a DSS shown in Figure 4.1, the items that are expected to be impacted by fuel burnup extending from 62 GWd/MTU to 85 GWd/MTU are:

- Principal Design Criteria Evaluation: Spent Fuel Design Basis
- Structural Evaluation: Component Materials
- Thermal Evaluation: Spent Fuel Cladding, Component Materials, Decay Heat
- Confinement Evaluation: Component Materials
- Shielding Evaluation: Component Materials
- Criticality Evaluation: Fissile Content Materials, Component Materials
- Materials Evaluation: Cladding Integrity.

Based on these needs identified and discussions in NUREG-2224, the following information on high burnup fuel and cladding is needed to support the safety analysis of a DSS containing high burnup fuel:

- New cladding material properties (yield stress, ultimate tensile strength, uniform elongation)
- Decay heat from high burnup fuel
- Fissile content.

In addition, codes and methods discussed in Section 3.0 will be necessary to provide bounding conditions regarding the following conditions of high burnup fuel:

- Rod internal pressure (likely will go up relative to bounding pressures discussed in NUREG-2224 because of enhanced fission gas release observed in high burnup fuel)
- Oxide thickness (should be specific for the alloy in question)
- Hydrogen content (should be specific for the alloy in question).

Finally, the radioactive source term should be re-evaluated for high burnup fuel. NUREG-2224 provides a table of fractions of fuel rods assumed to fail and radioactive fractions available for release for high burnup fuel in non-leak tight dry storage system designs, per ANSI N14.5 (American National Standards Institute, 2007). This table is reproduced as Table 4.1. This table is likely not applicable to burnup between 62 GWd/MTU and 85 GWd/MTU and should be reassessed based on the formation of the high burnup fuel rim that may lead to enhanced pellet fragmentation (see Sections 3.1.11 and 3.3.3.3) and the dramatic increase in fission gas release (see Section 3.1.8).

Table 4.1. Fractions of radioactive materials available for release from high burnup (up to 62 GWd/MTU) SNF under conditions of dry storage (for both PWR and BWR Fuels) (Ahn, et al., 2018)

Variable	Normal Conditions	Off-Normal Conditions	Accident-Fire Conditions	Accident-Impact Conditions
Fraction of Fuel Rods Assumed to Fail	0.01	0.1	1.0	1.0
Fraction of Fission Gases Released Due to a Cladding Breach	0.15	0.15	0.15	0.35
Fraction of Volatiles Released Due to a Cladding Breach	3×10^{-5}	3×10^{-5}	3×10^{-5}	3×10^{-5}
Mass Fraction of Fuel Released as Fines Due to a Cladding Breach	3×10^{-5}	3×10^{-5}	3×10^{-3}	3×10^{-5}
Fraction of CRUD Spalling Off Cladding	0.15	0.15	1.0	1.0

To address the issues of cladding creep and hydride reorientation, the applicant should also justify the following limits that are articulated in ISG-11 Rev. 3 and have historically been used for storage of SNF:

- 400°C peak cladding temperature limit for NCS
- Maximum cladding temperature limit of 570°C for off-normal and accident conditions
- The cladding should not experience more than ten thermal cycles each not exceeding 65°C.

Data that could be used to assess these needs will be summarized in Section 4.4.

4.3 Transportation of Spent Nuclear Fuel

This section discusses the transportation of SNF, including the current regulatory framework and its application to burnup between 62 and 85 GWd/MTU.

4.3.1 Current Regulatory Framework

The regulations in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” (US Nuclear Regulatory Commission, 2015) include a number of fuel-specific and package-specific requirements that may be dependent of the design basis condition of the fuel cladding. 10 CFR 71.31, “Contents of Application,” and 10 CFR 71.33, “Package description,” require an application for a transportation package to describe the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package, which includes a description of the chemical and physical form of the allowable contents.

The condition of the SNF cladding is critical to the transportation as 10 CFR 71.55(d)(2) requires that the geometric form of the package contents is not substantially altered under the tests for normal conditions of transport (NCT). 10 CFR 71.55(e) also requires that a package used for the shipment of fissile material is to be designed and constructed and its contents so limited that under the tests for hypothetical accident conditions (HAC) specified in 10 CFR 71.73, “Hypothetical Accident Conditions,” the package remains subcritical. The requirement assumes that the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents as stated in 10 CFR 71.55(e)(1).

In addition to these regulations, the NRC staff have provided NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel” (US Nuclear Regulatory Commission, 2000). This standard review plan provides guidance in the preparation by the applicant and review by the staff of a topical report describing a transportation package for SNF. The standard review plan lays out the following evaluations that should be performed during the safety evaluation of a DSS shown in Figure 4.2. Those areas impacted by high burnup are discussed in Section 4.3.2.

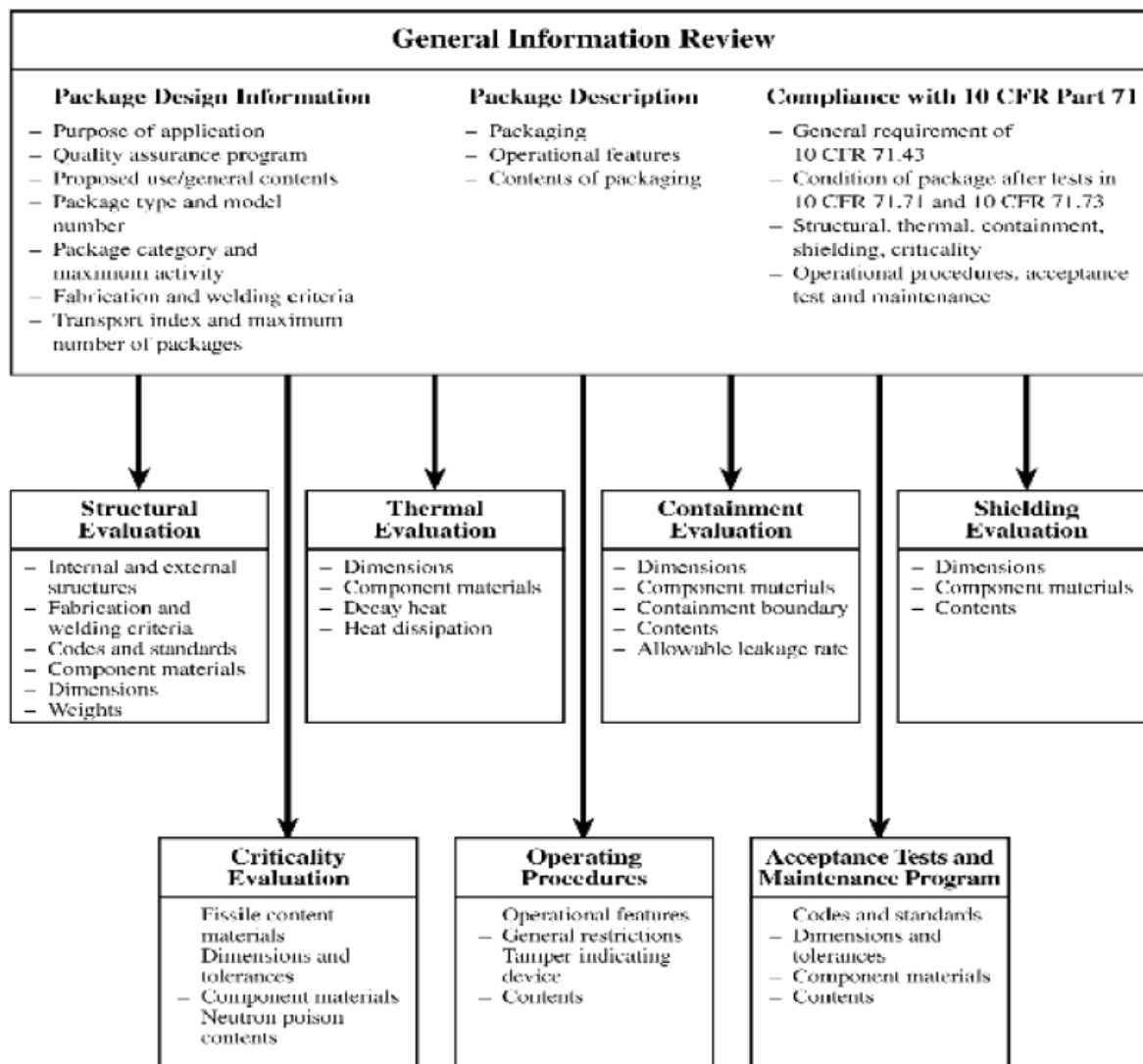


Figure 4.2. Overview of safety evaluation of SNF transportation (Taken from NUREG-1617)

As burnups have increased and new data have become available, NRC staff published Interim Staff Guidance 11 (ISG-11) regarding cladding considerations for the transportation and storage of spent fuel. This information is used to supplement the guidance in the standard review plan. The standard review plan (NUREG-1617) has not been revised to reflect any of this guidance. ISG-11 has been periodically revised as more data have become available. The summary of revisions are provided in Section 4.2.1

NRC staff has recently provided a draft NUREG that is out for comment (NUREG-2224 (Ahn, et al., 2018)) regarding dry storage and transportation of high burnup SNF. This document seems to consider fuel with burnup up to 65 GWd/MTU. Additionally, NRC staff recently presented their thoughts on the management of high burnup spent fuel (Torres, 2018). With regard to transportation of SNF, these documents discussed the roles of cladding fatigue lifetime and hydride reorientation. These documents also provided guidance for transportation of SNF that has been in dry storage for less than 20 years and for transportation of SNF that has been in dry storage for greater than 20 years.

4.3.2 Application to Burnup Between 62 and 85 GWd/MTU

In examining the safety evaluations that should be performed on a transportation package for SNF shown in Figure 4.2, the items that are expected to be impacted by fuel burnup extending from 62 GWd/MTU to 85 GWd/MTU are:

- Structural Evaluation: Component Materials
- Thermal Evaluation: Dimensions, Component Materials, Decay Heat
- Confinement Evaluation: Component Materials
- Shielding Evaluation: Component Materials
- Criticality Evaluation: Fissile Content Materials, Component Materials.

Based on these needs identified and discussions in NUREG-2224, the following information on high burnup fuel and cladding is needed to support the safety analysis of a transportation package for SNF containing high burnup fuel:

- New cladding material properties (yield stress, ultimate tensile strength, uniform elongation)
- Cladding fatigue lifetime
- Decay heat from high burnup fuel
- Fissile Content.

In addition, codes and methods discussed in Section 3.0 will be necessary to provide bounding conditions regarding the following conditions of high burnup fuel.

- Rod internal pressure (likely will go up relative to bounding pressures discussed in NUREG-2224 because of enhanced fission gas release observed in high burnup fuel)
- Oxide thickness (should be specific for the alloy in question)
- Hydrogen content (should be specific for the alloy in question).

NUREG-2224 concludes that the use of best-estimate cladding mechanical properties, not accounting for the presence of the fuel pellet, continue to be adequate for assessing the structural performance of high burnup SNF during the hypothetical 9 m and 0.3 m drops. Additionally, the hydride orientation is not a critical consideration when evaluating these cladding mechanical properties. PNNL assesses that this will continue to be the case for SNF with burnup between 62 and 85 GWd/MTU.

Finally the radioactive source term should be re-evaluated for high burnup fuel. NUREG-2224 provides a table of fractions of fuel rods assumed to fail and radioactive fractions available for release for high burnup fuel in non-leak tight transportation packages, per ANSI N14.5. This table is reproduced as Table 4.2. This table is likely not applicable to burnup between 62 GWd/MTU and 85 GWd/MTU and should be reassessed based on the formation of the high burnup fuel rim that may lead to enhanced pellet fragmentation (see Sections 3.1.11 and 3.3.3.3) and the dramatic increase in fission gas release (see Section 3.1.8).

Table 4.2. Fractions of radioactive materials available for release from high burnup (up to 62 GWd/MTU) SNF under conditions of transport (for both PWR and BWR fuels) (Ahn, et al., 2018)

Variable	NCT	HAC-Fire Conditions	HAC-Impact Conditions
Fraction of Fuel Rods Assumed to Fail	0.03	1.0	1.0
Fraction of Fission Gases Released Due to a Cladding Breach	0.15	0.15	0.35
Fraction of Volatiles Released Due to a Cladding Breach	3×10^{-5}	3×10^{-5}	3×10^{-5}
Mass Fraction of Fuel Released as Fines Due to a Cladding Breach	3×10^{-5}	3×10^{-3}	3×10^{-5}
Fraction of CRUD Spalling Off Cladding	0.15	1.0	1.0

To address the issues of cladding creep and hydride reorientation, the applicant should also justify the following limits that are articulated in ISG-11 Rev. 3 and have historically been used for transportation of SNF.

- 400°C peak cladding temperature limit for NCT.
- Maximum cladding temperature limit of 570°C for off-normal and accident conditions.
- The cladding should not experience more than ten thermal cycles each not exceeding 65°C.

The source of information that could be used to assess these needs will be summarized in Section 4.4.

4.4 Data Recommendation for Safety Evaluations

Based on the assessments in the previous sections, the information in Table 4.3 is needed to support safety analyses of a DSS and a SNF transportation package containing fuel with burnup between 62 and 85 GWd/MTU. Also shown in this table are the recommended sources of information for each of these items.

Some data necessary to support this information may already be available. Section 5.0 will discuss high burnup data that is currently available that could support the safety analysis of high burnup for reactor operation, SNF storage, and SNF transportation.

Table 4.3. Assessment data that could be used to justify the safety evaluation of a DSS and a SNF transportation package containing fuel with burnup between 62 and 85 GWd/MTU

Supporting Information	Recommended Source
<p>Mechanical properties (yield stress, ultimate tensile strength, uniform elongation)</p> <p>Separate effects tests to identify phenomena that can lead to gross cladding rupture</p>	<p>Mechanical property tests performed on cladding segments irradiated to high burnup. Note: mechanical properties used to support in-reactor analysis are typically performed at room temperature and reactor operating temperature (300°C to 350°C). These data should be collected at relevant temperature to storage and transportation.</p> <p>Failure limits at high burnup should be confirmed for creep strain capacity and delayed hydride cracking.</p>
Fatigue life	Fatigue tests performed on cladding segments that contain fuel or have been de-fueled irradiated to high burnup
<p>Justification for peak cladding temperature limits regarding hydride reorientation:</p> <ul style="list-style-type: none"> • 400°C peak cladding temperature limit for NCS and NCT • Maximum cladding temperature limit of 570°C for off-normal and accident conditions • The cladding should not experience more than ten thermal cycles each not exceeding 65°C 	Hydride reorientation and strength tests on cladding segments irradiated to high burnup
Limiting rod internal pressure, oxide thickness, and hydrogen content	Thermal-mechanical code approved to high burnup for rod with limiting power history
Source Term	Assessment of maximum fission gas release, volatiles, and fuel dispersal from high burnup fuel
Fractions of fuel assumed to fail under various conditions	Statistical assessment of high burnup fuel rods expected to be close to a cladding failure limit for each scenario
Decay heat	Standard method approved to high burnup such as ANS 5.1
Fissile content	Code prediction from code such as ORIGEN

5.0 Currently Available Data

This section contains a summary of the data that is currently available to support safety analyses of high burnup fuel (62-85 GWd/MTU). This summary contains those data that are currently publicly available. More proprietary data may exist, but these data will not be included in this summary. Most of the data included in this summary are those used in the assessment of FRAPCON (Geelhood & Luscher, 2015) and FRAPTRAN (Geelhood & Luscher, FRAPTRAN-2.0: Integral Assessment, 2016).

5.1 In-Reactor Data

This section discusses in-reactor data including fuel temperature, power ramp tests, and RIA tests.

5.1.1 Fuel Temperature

The in-reactor data that is most critical to the assessment of codes and methods used to perform safety analyses for high burnup fuel is fuel temperature data that is collected in-reactor during irradiation. The majority of this in-reactor temperature data was collected in the Halden reactor. Some notable high burnup temperature data include:

- IFA-562 Rod 18 – UO₂ irradiated to 76 GWd/MTU (Wiesenack, 1992)
- IFA-597 Rod 8 – UO₂ irradiated to 71 GWd/MTU (Matsson & Turnbull, 1998)
- IFA-515.10 – UO₂ and UO₂-Gd₂O₃ irradiated to 80 GWd/MTU (Tvergerg & Amaya, 2001).

Figure 5.1 shows the rod-average power and burnup range of the FRAPCON temperature assessment cases. It can be seen from this figure that there is a significant quantity of data up to 70 GWd/MTU and substantially less data between 70 and 85 GWd/MTU. Additionally, the power level for the data beyond 70 GWd/MTU is rather low.

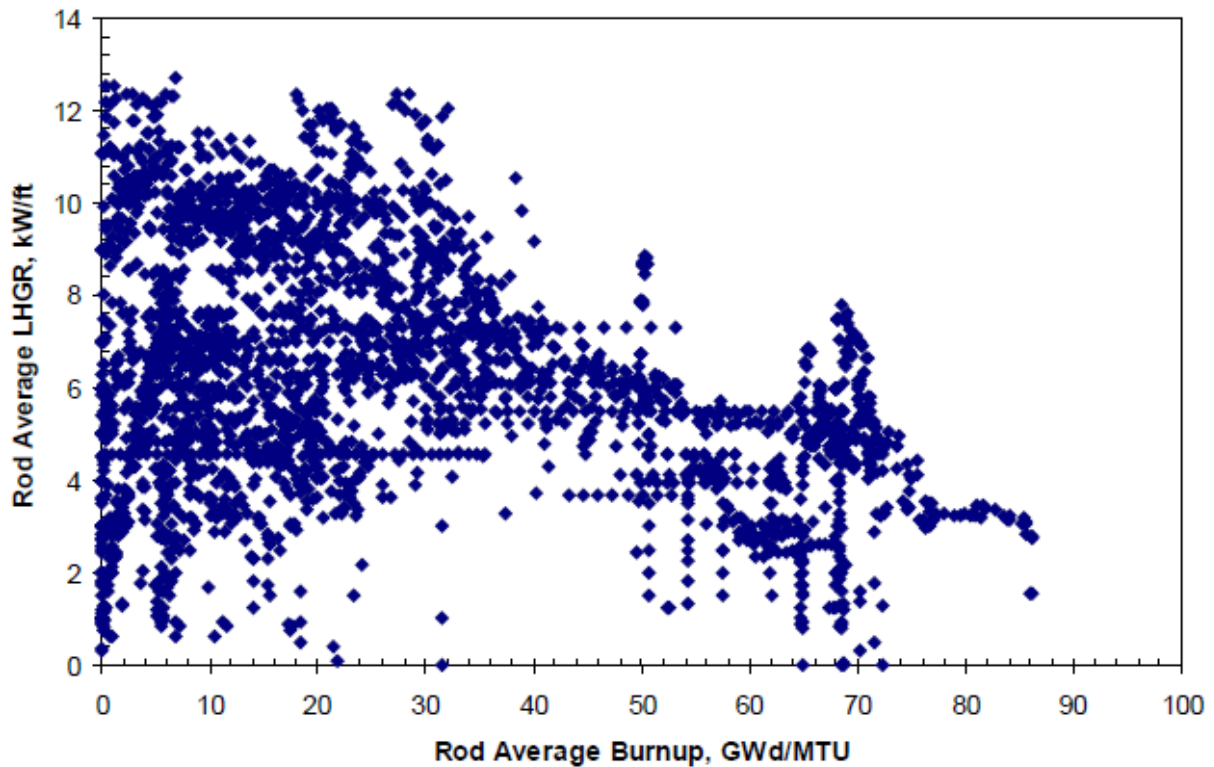


Figure 5.1. Rod-average LHGR vs. rod-average burnup for FRAPCON temperature assessment cases

5.1.2 Power Ramp Tests

Power ramp tests are performed to assess the cladding strain level that is induced by a rapid increase in power. Ramp tests are performed on refabricated fuel segments that have previously been irradiated to some burnup level. The permanent hoop strain is measured after the ramp test. During some power ramps, cladding failure is observed and these observations can be used to deduce a failure limit. Fission gas that is released during the ramp is also useful for assessing the power ramp fission gas release model. Some notable high burnup temperature data include:

- B&W Studsvik power ramps 62 GWd/MTU (Wesley, Mori, & Inoue, 1994)
- SCIP-1 Ramps to 72 GWd/MTU. (Kallstrom, 2005)

Figure 5.2 shows the rod-average power and burnup range of the FRAPCON hoop strain assessment cases. It can be seen from this figure that there is a significant quantity of data up to 72 GWd/MTU and no data beyond this level.

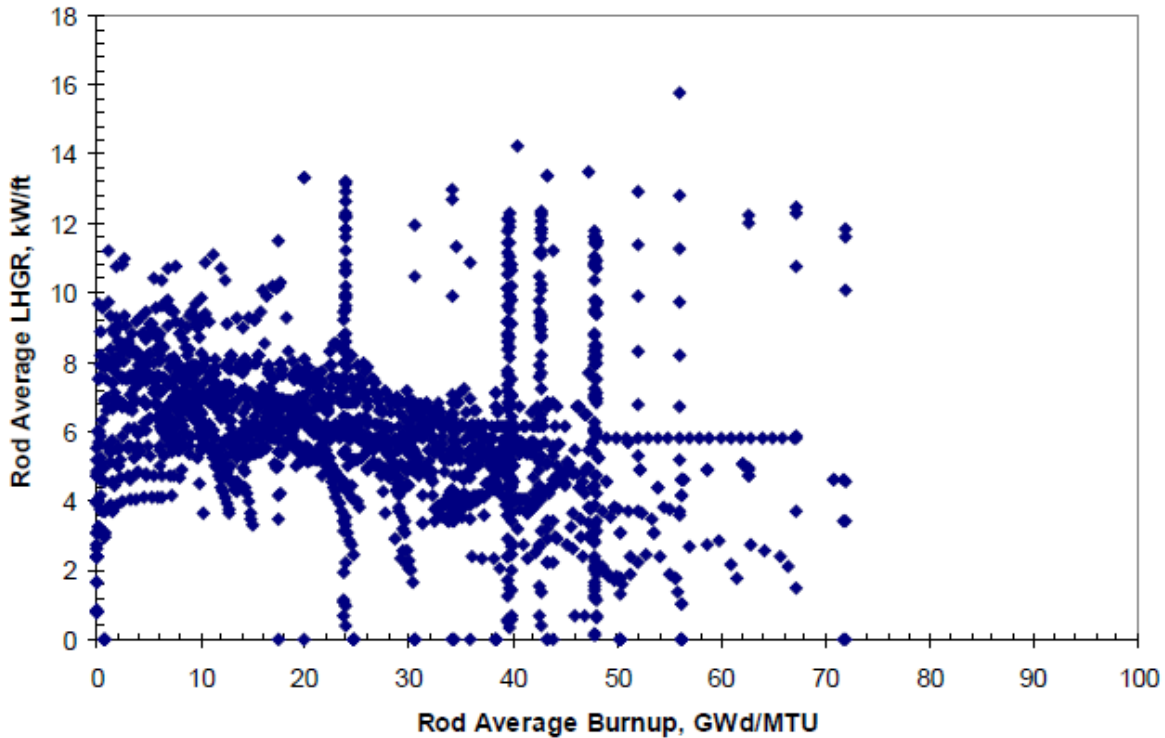


Figure 5.2. Rod-average LHGR vs. rod-average burnup for FRAPCON hoop strain assessment cases

5.1.3 RIA Tests

RIA prompt power pulse tests are performed to (1) derive safety criteria on total deposited energy which limit fuel rod damage (to preserve a coolable geometry) and molten fuel-coolant interaction and (2) develop cladding failure thresholds. RIA tests are performed on refabricated fuel segments that have previously been irradiated to some burnup level. Safety criteria may be sensitive to fuel composition (e.g., UO₂ versus MOX) and are burnup-dependent. Cladding failure thresholds may be sensitive to alloy microstructure (e.g., RXA versus SRA) and are presented as a function of cladding hydrogen content and rod internal pressure. Limited test data is available on fuel rod segments above 75 GWd/MTU with commercial cladding alloys.

Cabri and NSRR have tests up to 75 GWd/MTU (Cabri) and 80 GWd/MTU (NSRR) (Geelhood & Luscher, FRAPTRAN-2.0: Integral Assessment, 2016) (Mihara, Udagawa, & Amaya, Experimental Capability of Nuclear Safety Research Reactor (NSRR), 2017). The data up to 80 GWd/MTU seems sufficient to assess the failure limit up to this burnup level.

5.2 Ex-Reactor Data Taken on Irradiated Rods

This section discusses ex-reactor data for irradiated rods. This data includes fission gas release, cladding corrosion and hydriding, cladding mechanical properties, and integral LOCA tests.

5.2.1 Fission Gas Release

Fission gas release data is collected after irradiation of a fuel rod. This end-of-life fission gas release fraction is an integral measurement of the release that has taken place throughout the life of the rod. For this reason, it is important to know the power history for the rod with a puncture measurement. Low power rods are observed to release very little fission gas so they are not of importance in the assessment of a fission gas release model. It is desirable to have higher power rods that release a significant quantity of fission gas as these are the rods that will drive the pressure limits.

Figure 5.3 shows the rod-average power and burnup range of the FRAPCON fission gas release assessment cases. It can be seen from this figure that there is a significant quantity of data up to 70 GWd/MTU. There is a moderate level of data up to 100 GWd/MTU.

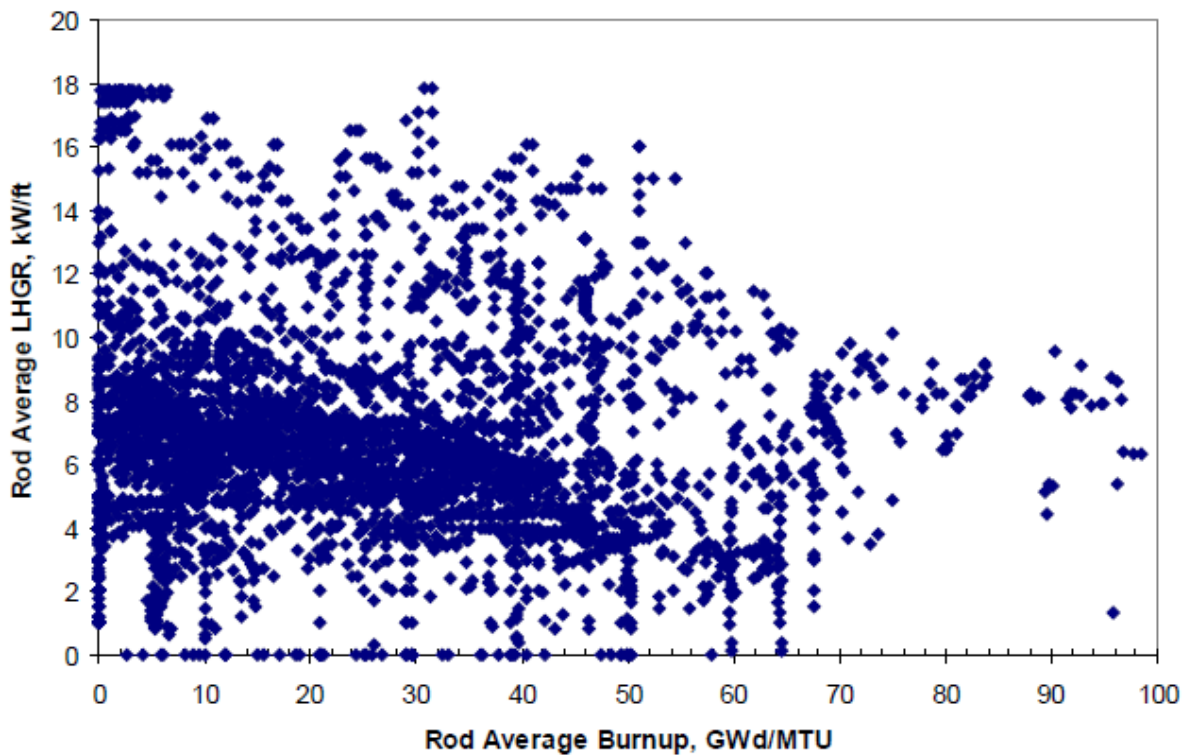


Figure 5.3. Rod-average LHGR vs. rod-average burnup for FRAPCON fission gas release assessment cases

5.2.2 Cladding Corrosion and Hydriding

FRAPCON has been assessed against some ZIRLO™ and M5® cladding up to 70 GWd/MTU (Geelhood & Luscher, 2015). There is sparse advanced cladding alloy corrosion data with a power history available in the public domain.

Applicants should provide significant corrosion and hydrogen content data and model predictions to justify high burnup fuel operation.

5.2.3 Cladding Mechanical Properties

Cladding mechanical properties are important for both the in-reactor and ex-reactor safety analyses of high burnup fuel. PNNL has collected a database of mechanical properties (Geelhood, Beyer, & Luscher, 2008). The report only addresses Zircaloy-2 and Zircaloy-4, however, PNNL does have M5® and ZIRLO™ data that shows the mechanical properties of these alloys does not significantly deviate from those of Zircaloy-2 and Zircaloy-4. Figure 5.4. shows the scope of the data. There is significant data up to 1.2×10^{26} n/m² (72 GWd/MTU) and sparse data beyond this up to 1.4×10^{26} n/m² (85 GWd/MTU).

Proprietary alloy-specific high burnup data should be used up to the requested burnup/fluence level to support licensing of high burnup fuel.

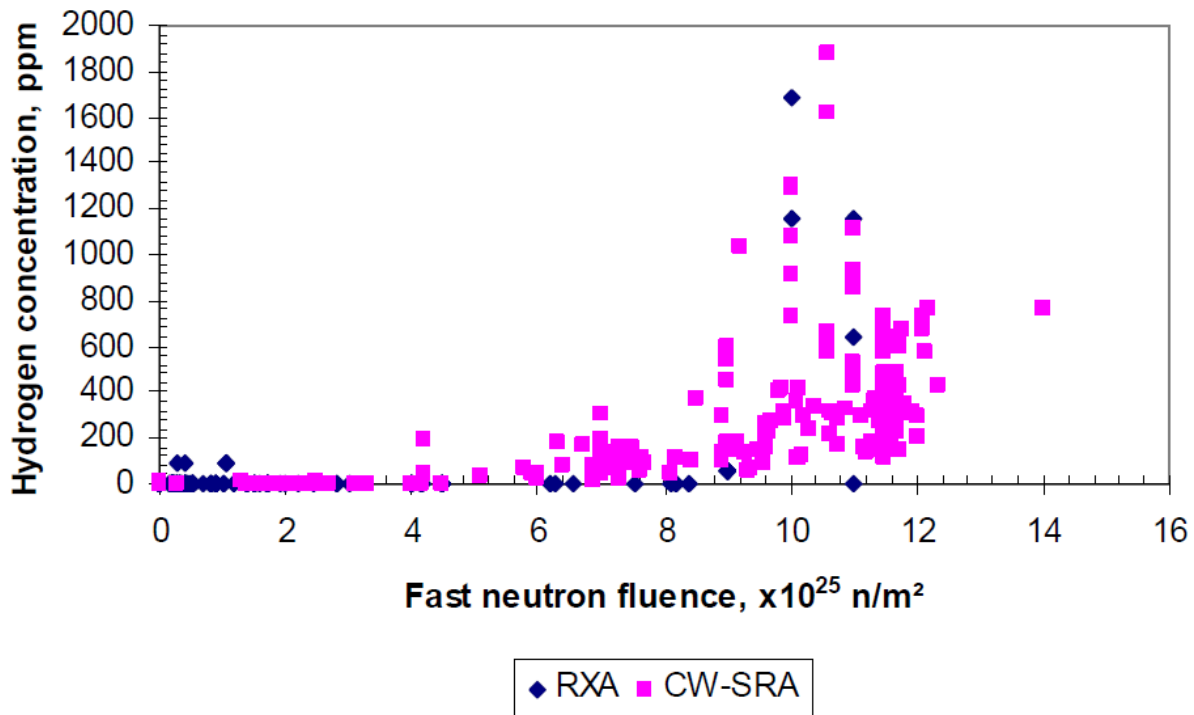


Figure 5.4. Hydrogen concentration vs. fast neutron fluence for the data in the PNNL database. (293K≤T≤755K)

5.2.4 Integral LOCA Tests

Integral LOCA tests have been performed in the Halden reactor for a number of fuel rods irradiated to 44 to 92 GWd/MTU. In these tests (Oberlander & Wiesenack, 2014) (Raynaud P. , 2012) it has been observed that there is some pellet fragmentation and relocation out of the burst opening for rods with burnup above 60 GWd/MTU and considerably more fragment dispersal above 80 GWd/MTU. This pellet release could challenge coolability limits following LOCA. This could also become an issue for RIA following cladding failure of high burnup fuel.

5.3 Data Gaps and Performance Concerns

As noted at the beginning of this section, the data compiled here are intended to give NRC staff the expected performance of high burnup fuel as well as areas of concern that should be given additional scrutiny during the review of a burnup extension request. The presence of data in any area does not indicate that an applicant would not have to provide data from their specific high burnup fuel design (fuel and cladding) because it has been observed that some cladding alloys have different performance at high burnup.

The following discussion summarizes at a high level the gaps in the publicly available data and any concerns about performance that the current data have brought to light.

5.3.1 Data Gaps

The following data needs are those where data is recommended for licensing, but currently limited publicly available data exists that would indicate with significant confidence what the performance of high burnup fuel would be. It is acknowledged that a majority of the available data are from small grain (10-15 μm) UO_2 rods. Fission gas release and cladding strain during power ramps may be sensitive to grain size.

- Fuel centerline temperature: significant data up to 70 GWd/MTU, sparse low power data up to 85 GWd/MTU.
- Power ramp tests: Significant data up to 72 GWd/MTU. No data above this level.
- RIA tests: Significant data up to 80 GWd/MTU. No data above this level.
- Fission gas release: Significant data up to 70 GWd/MTU. Moderate level of data up to 100 GWd/MTU.
- Cladding corrosion and hydriding: Significant data with power histories up to 70 GWd/MTU. Data with no power history up to 85 GWd/MTU.
- Cladding Mechanical properties: Significant data up to 72 GWd/MTU, Sparse data up to 85 GWd/MTU.
- Integral LOCA tests: Reasonable data between 44 and 92 GWd/MTU.

5.3.2 Performance Concerns

The following performance concerns for high burnup fuel have been identified based on the data presented in this section. Each is briefly discussed as follows.

Oxidation and Hydriding. As burnup progresses all cladding alloys will approach corrosion and hydrogen content limits. There is limited data from high burnup proprietary alloys that would indicate what burnup level these limits may be approached.

Strain and Strain Limits. Ductility goes down in high burnup fuel and gaseous swelling during power ramps could induce more strain in the cladding. It should be determined what the strain limit is for AOOs for the high burnup cladding in question based on mechanical property data. The code and methodology

should be assessed to ensure that it adequately predicts the strain during high-power ramp tests. Additionally, for storage of high burnup NSF, the creep strain limit is likely impacted by increased fluence.

Fission Gas Release and Rod Internal Pressure. The fission gas release is known to increase significantly at high burnup. The code and methodology should be assessed to ensure that fission gas release and rod internal pressure are adequately predicted at high burnup based on high burnup puncture data. Radioactive fission gas release modeling to high burnup is necessary to updated failed fuel source term assumptions.

Empirical Assembly Limits. Methods for examining spring relaxation and impact on hydraulic lift loads and fretting, rod bow, and lateral deflections should be examined relative to high burnup assembly performance to ensure that they remain adequate.

LOCA post quench ductility. LOCA post quench ductility may be reduced from the current Zr-alloy cladding embrittlement limits of 1204°C and 17% ECR at high burnup. Until the NRC revises these limits, the applicant should propose defensible limits.

RIA Cladding Failure. Limits should be proposed based on any relevant high burnup RIA tests and mechanical property data from high burnup cladding.

Fuel Fragmentation. Fuel fragmentation and dispersal has been observed in LOCA tests at high burnup. Pellet fragmentation is observed above 45 GWd/MTU and get progressively worse with increasing burnup. The expected quantity of fuel dispersal and its impact on retention of coolable geometry should be assessed for all design basis events.

Dry Storage and Transportation of SNF. As mentioned, for dry storage and transportation of SNF, the procedures and packages are designed around the fuel. Whoever, the phenomena that could occur in dry storage/transport may be different at higher burnups. As mentioned, for dry storage and transportation of SNF, the procedures and packages are designed around the fuel. Relevant high burnup fuel and cladding properties should be obtained to perform the required safety analyses. Separate effects testing is also needed to identify any changes to limits on cladding creep strain, or other failure mechanisms such as delayed hydride cracking.

6.0 Conclusions

The NRC is preparing for anticipated licensing applications requesting burnup extension for LWR fuel in United States commercial power reactors. Current burnup limits vary between the fuel vendors, but are at either 62 GWd/MTU rod-average burnup or 70 GWd/MTU peak pellet burnup. These two limits are essentially the same given how LWR fuel is operated. These anticipated applications are driving a need for regulatory preparedness by the NRC to receive LTR requesting approval for burnup extension between a rod-average burnup of 62 to 85 GWd/MTU.

Although it is the responsibility of the applicant to provide updated codes, methods, design limits, and the data to justify these, it is critical that NRC staff have an independent understanding of the existing data and potential damage mechanisms as well as the data that have previously been used to approve past burnup extensions.

This report provides current state of the industry information on material properties and fuel performance considerations for high burnup fuels between 62 and 85 GWd/MTU rod-average burnup in operating reactor conditions, reactor design basis accident conditions, SNF dry storage, and SNF transportation. To support the agency's efforts, this report identifies and discusses degradation and failure modes seen at high burnup including fuel performance characteristics of high burnup fuel that may not be addressed within existing regulatory documents (e.g., 10 CFR, regulatory guidance, standard review plans). The scope of this report includes high burnup LWR fuel with Zr-alloy cladding and UO₂ fuel. This will also apply to fuel with various burnable absorbers including UO₂-Gd₂O₃, UO₂-Er₂O₃, and IFBA rods (UO₂ with ZrB₂ coating).

Section 3.0 discusses changes to safety analysis codes, methods, and design limits for in-reactor performance in the context of high burnup fuel for both normal conditions and accident conditions. This section also identifies any potential new damage mechanisms that manifest at high burnup. Key items that have been identified as having greater uncertainty at high burnup and should be supported by test data are:

- Cladding oxide thickness and hydrogen concentration at high burnup
- Cladding strain limits at high burnup given increased cladding embrittlement
- Cladding axial growth at high burnup
- Enhanced fission gas release and rod internal pressure at high burnup
- Radioactive fission gas release to support accident source term tables
- Empirically derived limits
 - Spring relaxation and impact on hydraulic lift loads and fretting
 - Rod bow
 - Lateral deflections
- Accident analysis
 - Cladding embrittlement during LOCA

- Cladding failure during RIA
- Fuel dispersal during all design basis events.

Section 4.0 discusses changes to safety analysis codes, methods, and design limits for dry storage and transportation of high burnup SNF. Key items that have been identified as having greater uncertainty at high burnup and should be supported by test data are:

- Fatigue
- Cladding failure limits (creep strain and others)
- Fraction of rods assumed to fail and radioactive source term from failed rods (See Table 4.1 and Table 4.2)
- Empirically derived limits (temperature and cycling limits)
 - 400°C peak cladding temperature limit for NCS and NCT
 - Maximum cladding temperature limit of 570°C for off-normal and accident conditions
 - The cladding should not experience more than ten thermal cycles each not exceeding 65°C.

Finally, a discussion of the current out-of-pile and in-pile data has been performed in Section 5.0 with special consideration to availability or lack of data to support the damage mechanisms and performance considerations previously identified. This section also identifies performance concerns that have been identified for high burnup fuel.

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