INL/RPT-24-80116
Revision 0

Silicon Carbide Composite Cladding Qualification Test Plan

SEPTEMBER 2024

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Battelle Energy Alliance manages INL for the U.S. Department of Energy's Office of Nuclear Energy

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September 2024

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http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

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ABSTRACT

This report examines the process to qualify a new fuel-cladding type focusing on the data requirements and types of testing necessary to meet the regulatory burden and related guidance applicable to fuel qualification of accident-tolerant fuel. The report considers the use of accelerated fuel qualification techniques and lead-test specimen programs that may shorten the timeline for qualifying fuel for use in a nuclear reactor at the desired burnup. The assessment framework particularly emphasizes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of the experimental data used to develop and validate the empirical models and empirical safety criteria.

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ACRONYMS

AFQ	Accelerated Fuel Qualification
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATF	accident-tolerant fuel
CFR	U.S. Code of Federal Regulations
CHF	critical heat flux
CRDA	control rod drop accident
CREA	control rod-ejection accident
CRUD	Chalk River unidentified deposit
DBA	design-basis accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
ECCS	emergency core cooling system
EM	evaluation model
GDC	General Design Criteria
INL	Idaho National Laboratory
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LOFA	loss-of-flow accident
LOOP	loss-of-offsite power
LTA	lead-test assembly
LTR	lead-test rod
LWR	light water reactor
MCPR	minimum critical power ratio
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
PCI	pellet/clad interaction

PCMI	pellet/clad mechanical interaction
PWR	pressurized water reactor
RG	Regulatory Guide
RIA	reactivity-insertion accident
RIP	rod internal pressure
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
SAFDL	specified acceptable fuel design limit
SAR	Safety Analysis Report
SDM	shutdown margin
SiC	silicon carbide
SRP	Standard Review Plan
SSC	structure, system, and component
U.S.	United States
UO_2	uranium dioxide
Zr	zirconium

Silicon Carbide Composite Cladding Qualification

1. INTRODUCTION

Currently licensed fuel design in the United States (U.S.) light water reactor (LWR) fleet uses uranium dioxide (UO_2) fuel, enclosed within a zirconium (Zr)-alloy cladding. Years of research, accompanied by over 60 years of reactor operational experience at Idaho National Laboratory (INL) and other locations, have steadily generated technological advancements as well as an extensive experience base of both material response and performance in commercial- and test-reactor data for steady-state and transient conditions. However, events at the Fukushima Daiichi nuclear power generating station, coupled with recent material advancements, have provided a strong motivation for the industry and regulatory agencies to develop new fuel designs with an emphasis on improving performance, safety, and reliability for all operating and accident scenarios.

Accident-tolerant fuels (ATFs) are designed to increase coping time following a beyond-design-basis accident (BDBA) scenario while meeting all current technical licensing/regulatory requirements for design-basis accident (DBA) conditions—such as loss-of-coolant accidents (LOCAs) and reactivity-insertion accidents (RIAs)—while preserving or improving steady-state reactor operational performance. To demonstrate an improved fuel response, the advanced-fuel concept may need to reduce cladding oxidation-reaction kinetics, minimize the hydrogen-generation rate, improve the thermomechanical properties of the cladding, reduce pellet/cladding interactions, and/or reduce fission-product release.

This report considers the use of accelerated fuel qualification (AFQ) techniques and lead-test specimen programs that may shorten the timeline for generating the test data needed to qualifying fuel for use in a nuclear reactor at the desired burnup. The cladding of interest for this report is Silicon carbide (SiC). Silicon carbide has been investigated for use in both fission and fusion applications and has been considered as a candidate material for ATF cladding for LWRs. High-purity crystalline SiC is a very stable material under neutron irradiation, undergoing only minimal swelling and strength changes to 40 dpa and higher, which represents many times the exposure for a typical LWR fuel life. In addition, SiC retains its mechanical properties at high temperatures and reacts more slowly with steam than with Zr-alloys. However, monolithic SiC alone has low fracture toughness, making it unsuitable for nuclear-cladding applications where fuel containment is essential and a coolable geometry must be maintained, especially under transient or off-normal conditions. The solution is to employ an engineered composite structure to address this brittleness, using strong SiC fibers that reinforce a SiC matrix to form a SiC/SiC-composite. Compared to monolithic SiC, these composites offer improved fracture toughness and pseudo-ductility (OECD, 2012).

The properties of SiC-based cladding are highly dependent on the processing route used, particularly for any fiber-reinforced composite layers. In addition, while SiC/SiC-composites undergo pseudo-ductile fracture, rather than brittle failure, extensive microcracking occurs during this process. This can lead to a loss of hermeticity. This microcracking occurs at strains in the range of 0.1%, a strain level before which Zr-alloy cladding would not exhibit any plastic deformation (OECD, 2012). Accordingly, attention to characterization and careful development of test data are needed to demonstrate the mitigation of microcracking and ensure hermeticity through the operational life of the cladding.

2. FUEL QUALIFICATION

The term "fuel qualification" is not explicitly defined or used in U.S. Nuclear Regulatory Commission (NRC) regulations. However, there are regulatory requirements applicable to reactor applications that are generally associated with nuclear fuel behavior under conditions of normal operation, anticipated operational occurrence (AOOs), and accident conditions. The information needed to understand fuel behavior under these conditions is generally obtained as a part of a fuel qualification process. Research literature provides additional insight into the objectives of fuel qualification (Crawford et al., 2007), (Terrani et al., 2020), which highlights the needs to: (1) fabricate a fuel product in accordance with a specification, (2) meet licensing safety-requirements, and (3) meet reliability needs. This report uses the following definitions of "qualified fuel" and "fuel qualification" as an aid in the development of the fuel qualification path.

"Qualified fuel" means fuel for which reasonable assurance exists that the fuel, fabricated in accordance with its specification, will perform as described in the safety analysis. "Fuel qualification" means the overall process (planning, testing, analysis, etc.) used to obtain the qualified fuel.

These definitions are used because they characterize information required by NRC regulations, as described in Section 4 of this report, and are consistent with NRC staff experience from licensing LWR fuel rod designs.

3. REGULATORY BASIS

Nuclear fuel qualification to support reactor licensing involves the development of a basis to satisfy the regulatory requirements that apply to nuclear reactors. The fuel qualification pathway provides a means to identify the safety criteria for the fuel rod. These criteria are then used to establish the performance criteria for some structures, systems, and components (SSCs) of the facility. Therefore, the overriding regulatory requirements are associated with and addressed by the descriptions and safety analysis of the affected SSCs.

The relevant regulatory requirements associated with fuel qualification are as follows:

- The regulation in 10 Code of Federal Regulations (CFR) 50.43(e)(1)(i) requires the demonstration of the performance of each safety feature of the design through either analysis, appropriate test programs, experience, or a combination thereof. The pathway developed in Section 4 of this report provides a means to identify the safety features of the fuel necessary to comply with regulatory requirements, discusses the types of data and analyses typically expected to demonstrate the identification, and assists in the understanding of fuel life-limiting failure and degradation mechanisms that are due to irradiation and exposure to the in-reactor environment.
- The regulation in 10 CFR 50.43(e)(1)(iii) requires that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.
- The regulations in 10 CFR 50.34(a)(1)(ii)(D), 10 CFR 52.47(a)(2)(iv), and 10 CFR 52.79(a)(1)(vi) require an evaluation of a postulated fission-product release.
- The regulations in 10 CFR 50.34(a)(3)(i), 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a) require that the principal design criteria (PDC) be provided for a construction permit, design certification, combined license, standard design approval, or manufacturing license. Appendix A to 10 CFR Part 50, "General Design Criteria (GDC) for Nuclear Power Plants," establishes the minimum requirements for PDC for water-cooled nuclear power plants (NPPs).

Accordingly, NRC staff expects that information be provided that address the design aspects described in the following GDC as part of the fuel qualification process:

- GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Appendix S to 10 CFR 50, "Earthquake Engineering Criteria for Nuclear Power Plants," implements GDC 2 as it pertains to seismic events and defines specific NPP earthquake criteria. This appendix established definitions for safe-shutdown earthquake (SSE), operating basis earthquake (OBE), and safety requirements for relevant SSCs. These SSCs are necessary to assure the integrity of the reactor coolant pressure boundary (RCPB), the capability to shut down the reactor and maintain it in a safe-shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures. The safety functions generally associated with nuclear fuel include control of reactivity, cooling of radioactive material, and confinement of radioactive material.
- GDC 10, "Reactor Design," requires that specified acceptable fuel design limits (SAFDLs) or specified acceptable radionuclide release design limits (SARRDLs) not be exceeded during any condition of normal operation, including the effects of AOOs.
- GDC 27, "Combined Reactivity Control Systems Capability," requires, in part, the ability to achieve and maintain safe shutdown under postulated accident conditions and provide assurance that the capability to cool the core is maintained.
- GDC 35, "Emergency Core Cooling," requires an emergency core cooling system (ECCS) that provides sufficient cooling under postulated accident conditions; this GDC also requires that fuel and clad damage that could interfere with continued effective core cooling be prevented.

In order to further define the applicable CFRs and GDC requirements, the NRC has prepared several guidance documents that are related to nuclear fuel qualification. The most important of which is NUREG-0800, Section 4.2, Revision 3, "Fuel-System Design," issued March 2007 (NRC, 2007). This document lists the acceptance criteria that the NRC staff consider in a licensing review for a LWR fuel system. These criteria are as follows:

- Assurance that the fuel system is not damaged as a result of normal operation and AOOs can be demonstrated.
- Assurance that fuel-system damage is never so severe as to prevent negative reactivity insertion (e.g., control element insertion) when required.
- Assurance that the number of fuel failures is not underestimated for postulated accidents can be demonstrated.
- Assurance that coolability is always maintained can be demonstrated.

Though neither the NRC regulations nor NUREG-0800 uses the term "fuel qualification," Section 4.2 provides acceptable criteria to support findings associated with fuel performance. It should also be noted that Section 4.2 provides guidance based on the performance history of Zr-based alloy cladding. LWR fuel and licensing bases for traditional LWR power plants was tailored for known failure mechanisms from traditional UO_2 fuel with Zr-alloy cladding. As such, specific acceptance criteria provided in Section 4.2 may not be applicable or may not sufficiently address failure mechanisms of SiC/SiC-composite material. However, it is important that the basic material properties and any potential new failure mechanisms be identified, and that sufficient experimental data be developed to support proposed alternative criteria to bound identified failure mechanisms.

4. NUREG-0800 REVIEW REQUIREMENTS

Experimental data will have to be developed in order to provide the basis of any licensing submittal for a new cladding system. As discussed in the Section 3 of this report, Section 4.2 of NUREG-0800 discusses the review criteria for nuclear fuel systems and provides an outline for the types of data needed for licensing. Although some of the criteria may not be directly applicable to SiC/SiC-composite cladding concepts, it does highlight the required phenomena that needs to be addressed by the test plan.

There are three categories of phenomena and data needs that are discussed in Section 4.2 to bound normal operations, AOOs, and DBAs. The goal is the identification of fuel and cladding material and behavioral data typically used to verify the impact of irradiation conditions on fuel phenomena and parameters affecting fuel-rod performance. The types of data that will be needed include thermal and mechanical properties, such as thermal conductivity, specific heat, melting temperature, swelling, creep and ductility as a function of temperature and fuel burnup. Also included are behavioral data, including fission gas release and irradiation growth/swelling. This type of data may come from different sources including separate-effects tests, special irradiation programs in test reactors, and material property tests. This data may be used to justify an existing criterion or justify revisions to the current technical licensing bases. These data may also be obtained from mechanical-property tests and poolside and/or destructive examinations. The fuel material used to obtain the required data may come from fuel assemblies operated under prototypical power and coolant chemistry conditions. Generally, these data may be reported in licensee fuel topical reports. Appendix A of this report provides a detailed description of all the current licensing criteria. A summary of the data needed to demonstrate compliance with the criteria is provided in the following tables.

The fuel-system damage criteria are intended to meet the requirements of GDC 10, as these requirements relate to SAFDLs for normal operation, including AOOs. The purpose of the criteria and data needed to demonstrate compliance listed in Table 1 is to assure that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Testing methods used to collect the data include single-effects tests and integral tests depending on the data being collected. Data should be collected on samples irradiated up to the planned operational burnup limits expected. Additionally, not all of these criteria may be suitable for SiC/SiC-composite cladding and alternative criteria can be proposed as an alternative.

Table 1. Fuel-system SAFDLs review.

Design-Basis Limits/Criteria	Purpose of Limit	Tests and Data to Demonstrate Compliance
Design Stress	Prevent failure of cladding from overstress conditions.	Yield stress/ultimate tensile strength measurements.
Design Strain	Prevent failure of cladding from excessive strain conditions.	Uniform elongation/total elongation measurements.
Design Fatigue	Prevent failure of cladding from cyclic fatigue.	The design limit for strain fatigue is based on the cumulative fatigue usage factor for the cladding. Most licensees specify that the cumulative (life) fatigue usage factor shall be less than 1.0, considering a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is more conservative. Other proposed limits may need to be justified.

Design-Basis Limits/Criteria	Purpose of Limit	Tests and Data to Demonstrate Compliance
Fretting	Excessive fretting wear can lead to failed cladding.	The use of data taken from operating reactors (experience) and out-of-reactor flow tests to provide assurance that failures are of a limited grid design.
Oxidation	Retain cladding ductility as stated in cladding strain limit.	Oxidation/erosion test data in a coolant chemistry similar to the expected operational environment.
Hydriding	Retain cladding ductility as stated in cladding strain limit.	Hydriding is not an expected issue for SiC/SiC-composite cladding.
Chalk River unidentified deposits (CRUD)	Retain cladding ductility as stated in cladding strain limit.	CRUD layer thickness generally increases with residence time in the reactor. CRUD inventories and deposition characteristics are more a function of reactor coolant cleanup system capabilities and water chemistry practices than simply burnup. In pressurized water reactors (PWRs), a key factor affecting CRUD deposition and build up is the Subcooled Nucleate Boiling (SNB) duty. This is also most significant at lower burnups.
Rod Bow	Could impact DNBR or MCPR.	Rod-to-rod gap closure measurements on irradiated fuel assemblies to ensure the empirical rod bow method appropriately models lead-test assembly (LTA) rods. besides the axial growth, the differential swelling across the cladding cross section can also cause rod bowing (Li et al., 2020).
Irradiation Growth	Excessive assembly growth could lead to assembly deformation.	Fuel rod/assembly length measurements on irradiated fuel assemblies. Unlike metal claddings, ‘irradiation growth’ is mostly due to irradiation induced swelling effects.
Internal Gas Pressure	Prevent significant deformation resulting in departure of nucleate boiling (DNB).	Integral rod internal gas pressure measurements.
Hydraulic Lift Loads	The weight of the assembly and force of hold-down springs should prevent assembly liftoff.	Hold-down spring-force measurements.
Control Rod Reactivity and Insertability	Lateral deflections should not be so great as to prevent control rods/blades from being inserted.	Fuel assembly lateral deflection measurements.

The fuel-rod failure section of NUREG-0800 Section 4.2 provides an overview of the technical fuel-rod failure criteria and applies to normal operation, AOOs, and DBAs. The design limits and data needs are summarized in Table 2. When the failure thresholds are applied to normal operation, including AOOs, they are used as SAFDLs since fuel failure under those conditions should not occur according to the traditional interpretation of 10 CFR 50, Appendix A, GDC 10. When these thresholds are used for DBAs, fuel failures are allowed but may need to be accounted for in the dose calculations required by 10 CFR 50.67 and 10 CFR 100. Fuel-rod failure is defined as the loss of fuel-rod hermeticity. When applicable, the fuel-rod failure criteria should consider high-burnup effects based on the irradiated material properties data.

Table 2. Fuel failure SAFDLs review.

Design-Basis Limits/Criteria	Purpose of Limit	Tests and Data to Demonstrate Compliance
Internal Hydriding	Retain cladding ductility as stated in cladding strain limit.	This is based on the moisture content of the fuel and has not been an issue with the adoption of modern manufacturing processes. Internal hydriding may need to be considered with the introduction of metal liner/liquid metal materials to the fuel-cladding gap.
Cladding Collapse	Prevent failure of cladding due to collapse in the plenum and axial pellet gaps which result in large local strains.	Fuel column axial gap measurements on integral test samples can be taken to demonstrate performance.
Overheating of Cladding	Failure of cladding and dose consequence if critical heat flux is exceeded.	Tests on unirradiated tubes to establish thermal limits (CHF) should be performed on prototypic tubes. Fuel-rod cladding temperature calculations.
Overheating of Fuel Pellets	Prevent fuel melting during LOCA to assure that axial or radial relocation of molten fuel would neither allow molten fuel to contact the cladding nor produce local hot spots. Melting should also be precluded during RIA to reduce violent expulsion of fuel.	Fuel centerline temperature data. Fuel centerline temperature calculations.
Excess Fuel Enthalpy	Failure of cladding and dose consequence during RIA if injected energy limit is exceeded. Two limits are in place regarding maximum fuel enthalpy to evaluate fuel failure and core cooling.	RIA tests or surrogates such as rapid heating and loading on irradiated fuel segments could be used to develop RIA failure criteria. Fuel enthalpy calculations.
Pellet/Cladding Interaction	Prevent cladding failure from chemically assisted cracking.	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. Uniform elongation/total elongation measurements. Will need to develop an alternative to the 1% strain criteria.
Clad Rupture	Bursting of the fuel rod relates to failure of fuel rods due to high temperature and high gas pressures	Cladding tube burst tests are not appropriate for SiC/SiC-composites due to the pseudo-ductile nature of the material. It is suggested

Design-Basis Limits/Criteria	Purpose of Limit	Tests and Data to Demonstrate Compliance
	during a LOCA. This can also be a consideration during RIA.	that using plug expansion tests as a replacement for burst tests. 3-pt./4-pt. bend tests can be used to define the nature of the failure mechanisms and to demonstrate that no fuel material will be lost.
Mechanical Fracturing	Failure of cladding and dose consequence from external event.	Data required includes stress at which micro-cracking begins and the ultimate tensile stress collected on irradiated test specimens.

The fuel coolability criteria applies to DBAs. Table 3 provides a summary list of the criteria and data typically used to justify the design bases to verify compliance. For accidents in which severe fuel damage might occur, core coolability must be maintained to meet the requirements of 10 CFR 50 Appendix A, and GDC 27 and 35 as they relate to control rod insertability and core coolability. Coolability, or a coolable geometry, has traditionally implied the fuel assembly retains its rod-bundle geometry with sufficient coolant channels to permit removal of residual heat.

Table 3. DBA SAFDLs review.

Design-Basis Limits/Criteria	Purpose of Limit	Tests and Data to Demonstrate Compliance
Cladding Embrittlement	Coolable geometry must be retained following LOCA. There should be no post-LOCA general fuel/assembly failure.	Cladding embrittlement is not an issue with SiC/SiC-composites. It may be more appropriate to use 3-pt./4-pt. bend tests to define the nature of the failure mechanisms and to demonstrate that no fuel material will be lost to the coolant. In addition, to develop more margin during a LOCA, it may be beneficial to develop a peak fuel temperature criterion as the current peak cladding temperature limit was determined based on accelerated oxidation kinetics.
Violent Expulsion of Fuel	Coolable geometry must be retained following an RIA. Resulting pressure pulse must not damage reactor vessel.	This was provided by Transient Reactor Test (TREAT) facility integral test data and the 3D neutronic calculations.
Generalized Clad Melting	Coolable geometry must be retained following LOCA.	The current limit is set as the cladding melting temperature. It should be noted that that SiC does not melt, it just sublimates at high temperatures ~2700C.
Fuel-Rod Ballooning	Degree of ballooning needed to calculate blockage of the coolant channel.	SiC/SiC-composite cladding will not balloon as seen in traditional Zr-alloy cladding. Tube burst tests can be conducted to verify no ballooning. To demonstrate the lack of large amounts of fuel debris escaping the pin and to show a coolable geometry is maintained, the bend test data can be used to show that failure does not lead to gross cladding deformation or catastrophic failure.

Structural Deformation	Coolable geometry must be retained following LOCA or seismic event.	This is more of an issue for assembly performance and is dictated by the grid design yield stress.

5. ACCELERATED FUEL QUALIFICATION

The AFQ process has been proposed to speed up the evolutions necessary to qualify a new fuel system for use in commercial NPPs. AFQ involves the use of advanced modeling and simulation to inform constituent and system selection and to enable integral fuel performance analyses (Terrani et al., 2020). The AFQ process, as shown in Figure 1, supports the identification of important parameters and phenomena for targeted characterization through separate-effects tests.

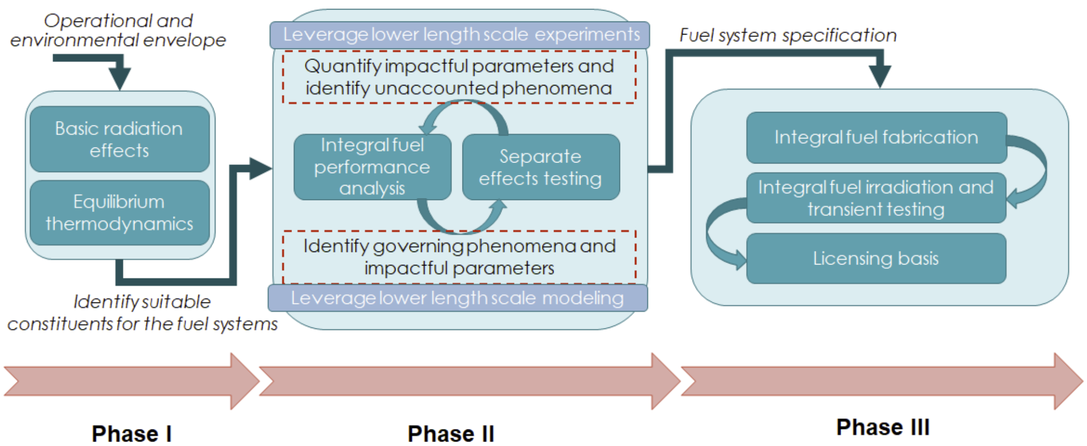


Figure 1. AFQ process workflow (Terrani, et al., 2020).

Advanced separate-effects testing techniques, such as fission-accelerated steady-state testing (FAST) (Beausoleil II, Povirk, & Curnutt, 2020) and MiniFuel (Petrie, Burns, Raftery, Nelson, & Terrani, 2019) can reduce the time needed to achieve a given burnup and provide basic data on material behavior and property evolution under irradiation conditions. The information obtained through these analyses and separate-effects tests help justify the adequacy of the EM. Additionally, validated physics-based models may support some extrapolation of EMs beyond the limits of the available integral test data. Ultimately, the AFQ process relies on integral irradiation test data to validate engineering-scale fuel performance codes and to confirm the performance and safety of the fuel system under prototypic conditions. Accordingly, the integral test data produced as part of the AFQ process appear to be consistent with the considerations in developing the experimental data assessment. Important aspects of AFQ include the use of fission accelerated testing with the desire to reduce the number of required integral irradiation tests.

5.1. Fission Accelerated Testing

Fission-accelerated test techniques, such as FAST and MiniFuel, can provide insight into the physics modeling needs for EMs without the need for large-scale integral testing. However, data used to assess EMs should be representative of the prototypical conditions. Test specimens should be fabricated in a manner that is consistent with fuel manufacturing specifications and any test distortions (e.g., differences in test dimensions or conditions) should be adequately justified.

Accelerated fuel irradiations may provide enhanced understanding of fuel behavior, but the NRC staff has not identified information describing how differences in test specimen manufacturing or test conditions impact the data obtained from these irradiations (i.e., potential biases or gaps in the data due to distortions in the test conditions). Accordingly, accelerated irradiation techniques can be a valuable tool for increasing the understanding of fuel behavior under irradiation, but additional justification will still be needed to justify the use of test data obtained from accelerated irradiation testing for assessing EMs in lieu of engineering-scale integral test data obtained under a quality assurance program.

5.2. Reduced Number of Integral Tests

AFQ seeks to reduce the number of required integral irradiation tests through the use of physics-based models instead of empirically based models (AFQ Working Group, 2021) (GA-EMS, 2021). This reduction in the number of required irradiation tests is because empirically based models require a much larger database to develop and train the empirical model. However, sufficient irradiation data is still needed to assess EMs. One possible benefit of using validated physics-based models is to support some extrapolation of EMs beyond the limits of available integral test data, while another potential benefit is to justify a reduced number of integral irradiation tests. This justification could come in the form of demonstrating an understanding of key degradation and performance phenomena through the validation of the physics-based models.

One potential challenge of using irradiated fuel samples is collecting sufficient data to establish confidence intervals for EM uncertainty. In scenarios where sufficient data is not collected, a more bounding or conservative approach also can be taken showing the model is inherently conservative or applying a bias or penalty to the model prediction.

5.3. Lead-Test Specimens

Much of the data necessary to qualify fuel for use comes from irradiated test specimens. Lead-test specimens have been successfully used in operating reactors to obtain data at the needed exposures and are discussed in NUREG-0800, Section 4.2.

5.4. First Core Applications

Nuclear fuel contributes to the reactivity balance and is a source of heat generation and fission products. Therefore, it is generally recognized as impacting the safety functions of reactivity control, heat removal, and confinement of radioactive material. Accordingly, nuclear reactor fuel is generally considered a safety feature subject to the requirements of 10 CFR 50.43(e) and would require demonstration prior to licensing. To address the regulatory requirements, sufficient data on the safety features of the fuel rod is necessary to support licensing. This data may be obtained from test reactors such as the Advanced Test Reactor (ATR) and TREAT located at INL that can be used for assessing the EMs.

6. OTHER CONSIDERATIONS

Another goal to consider while developing the test plan and experiment design is an assessment of expected fuel-cladding performance under postulated transient and accident conditions. This helps to bound the experiment parameters needed to generate data that captures the breath of the anticipated operational envelope. The applicable postulated events requiring analysis are specified in Section 15.0 of NUREG-0800 (NRC, 2007). For commercial LWR NPPs in the U.S., an analysis of the following design-basis accidents (DBAs) is required:

- A LOCA consisting of a rupture of a pipe containing reactor coolant up to and including the double-ended rupture of the largest pipe within the RCPB.
- Major rupture of a secondary system pipe up to and including a double-ended pipe rupture.
- An RIA consisting of the ejection of a PWR control rod assembly.
- A loss-of-flow accident (LOFA) resulting from a single reactor coolant pump locked rotor in a PWR.

In addition to these DBA events, the licensing of the fuel system for insertion into a PWR also may require an assessment of the some of the spectrum of AOOs identified in Section 15.0 NUREG-0800 (NRC, 2007). The list of these AOOs for PWRs is provided below:

- Loss/interruption of core coolant flow (excluding PWR reactor coolant pump locked rotor)
- Improper fuel assembly position
- Inadvertent control rod or rod group withdrawal
- Inadvertent moderator cooldown
- Inadvertent chemical shim
- Control rod drop (inadvertent addition of neutron absorber)
- Depressurization by spurious active element operation (e.g., relief valve)
- Reactor coolant blowdown through a safety relief valve (SRV)
- Loss of normal feedwater
- Loss of condenser cooling
- Steam generator tube leaks or rupture (SGTR)
- Reactor–turbine load mismatch (including load rejection and turbine trip events)
- Single operator error
- Single failure of core component
- Single electrical system failure
- Small reactor coolant system (RCS) system leak (small-line break or crack in large pipe)
- Minor secondary system break
- Loss-of-offsite power (LOOP)
- Loss-of-feedwater heating
- Reactor overpressure with delayed scram.

A necessary condition for the adoption of the cladding for commercial use is that its performance under normal operational conditions meets or exceeds the current Zr-based alloy/UO_2 fuel system. Since the basic structure of the fuel is not significantly different from fuel designs currently in use at operating NPPs (e.g., UO_2 ceramic fuel pellets), the technical criteria applied to current fuel designs are anticipated to remain mostly applicable for the licensing and operation of SiC/SiC cladding concepts. However, the area of licensing including setting appropriate operating thermal limits (e.g., linear heat generation rate [LHGR], departure from nucleate boiling ratio [DNBR]/ minimum critical power ratio [MCPR], etc.), reactivity and power distribution limits (shutdown margin [SDM], moderator temperature coefficient [MTC], heat flux hot channel factor [F_Q], enthalpy rise hot channel factor [$F_{\Delta H}$], etc.), and other limits will be affected by the cladding material performance.

In addition, there are a number of potential operational issues that are not related to licensing of the cladding but are important considerations for the decision on inserting the cladding into a commercial core. These issues include:

- Radiation dose implications (due to phenomena such as activation products of new ATF materials and tritium diffusion through the clad) on plant personnel and equipment.
- Impact of changes in neutron flux spectrum and reactivity coefficients on operational parameters (such as thermal limit margin, reactor kinetics, effectiveness of soluble boron, and control rod worth).

- Impact on operational parameters—in particular, operational and shutdown reactor water chemistry, CRUD, and corrosion (hydrogen water chemistry, injection of corrosion inhibitors, etc.).
- Impact on nuclear instrumentation, setpoints, and response time of automatic protection features (e.g., reactor scram setpoints, response time).
- Impact on operations procedures and related licensed reactor operator training and plant simulator models.
- Impact on spent fuel storage (both in the spent fuel pool and dry cask storage).
- Impact on the length or complexity of the refueling outages (e.g., changes to fuel handling, fuel shipping, decay heat).
- Impact to fuel examinations (fuel assembly growth, complexity of fuel handling and disassembly/reassembly).

Some of these issues may need to be evaluated (at least in a preliminary manner) prior to expending significant resources to support demonstration activities, such as inserting lead-test rods (LTRs) or LTAs. It is prudent to provide a catalog of these issues so they can be assessed in time to support these critical decisions.

One area for investigation related to SiC/SiC-composite cladding is the potential to replace the current departure from nucleate boiling (DNB) criteria with one that evaluates time at temperature under fuel dryout conditions. Due to the design and complex behavior of the fuel during NPP operation, specification of thermal design limits in terms of fuel structural design parameters (e.g., strain and fatigue limits) has been found to be impractical. Thus, for the licensing of fuels currently in use at operating NPPs, these limits are specified in terms of limits on material temperatures and heat flux conditions. One such parameter is the DNBR for PWR plants. These limits are specified as a relevant criterion in the assessment of fuel performance to ensure the fuel cladding does not exceed 1% strain during AOO conditions. Critical heat flux (CHF) (also referred to as the boiling crisis) corresponds to the condition where a phase change in the reactor coolant occurs during heating. In a PWR, CHF occurs when the bubble density from nucleate boiling is sufficient for adjacent bubbles to coalesce and form a vapor film on the surface of the fuel rod. Heat transfer under these conditions is relatively low, relative to the liquid phase, resulting in a marked increase in the cladding surface temperature. Thus, DNB has been used as a conservative criterion to ensure the boiling transition does not occur during normal operating conditions or AOOs.

It has been known for some time that exceeding the CHF for short periods of time would not necessarily produce adverse effects on fuel cladding. As a result, Japanese researchers have investigated the effects of post-transition boiling on fuel integrity with the objective of establishing criteria for the reuse of fuel in the event that it would experience an incident in which transition boiling occurred. This research identified that three critical correlations are needed to predict fuel-cladding temperatures during boiling transition events (Hara, 2003):

- The onset time of boiling transition
- The heat transfer coefficient between the cladding surface and the coolant
- Rewet time.

As a result of this research, the standards committee of the Atomic Energy Society of Japan (AESJ) proposed guidelines for judging boiling water reactor (BWR) fuel integrity after experiencing a boiling transition event and for reuse of these fuel assemblies (in Japan) using methods and time at temperature acceptance criteria as described in Hara (2003).

It has been known for some time that SiC/SiC-composite cladding will begin to microcrack at strains as low as 0.1%. Therefore, the 1% strain used currently is not an appropriate metric to use. Moving to a time at temperature acceptance criteria in place of the DNBR currently used is an important step for implementing the SiC/SiC-composite for commercial use.

Additional test work will need to be completed to demonstrate successful end-plug joining technology and the influence of irradiation swelling on deformation and bowing on irradiated prototypic cladding integral test specimens.

7. CLADDING QUALIFICATION PATHWAY

A path forward drawing on regulatory history as derived from the licensing of current fuel-cladding designs and AFQ considerations is shown in Figure 2.

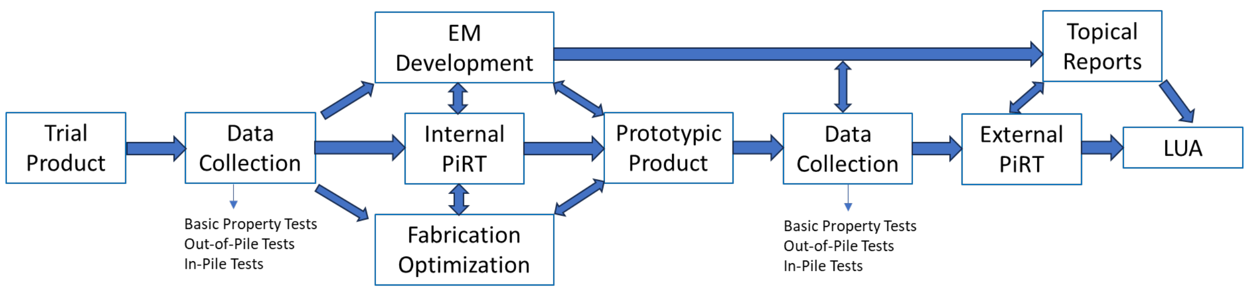


Figure 2. Path forward for cladding qualification.

The data collected through basic material property testing, single-effects tests, and integral tests in reactor tests in facilities such as the INL's ATR and TREAT facilities can be used to develop and validate the EMs and provide feedback into fabrication optimization. Data collected from experiments derived from the testing plan for the development of the prototypic cladding fabrication method can be used to train the EM. Once the prototypic cladding is available for irradiation, the experimental data needs to be focused to train any empirical methods in the EM and then validating the EM to the desired burnups. Table 4 shows some of the cladding material properties that are typically needed to perform fuel thermal-mechanical analysis of nuclear fuel with Zr-alloy cladding under normal conditions and AOOs.

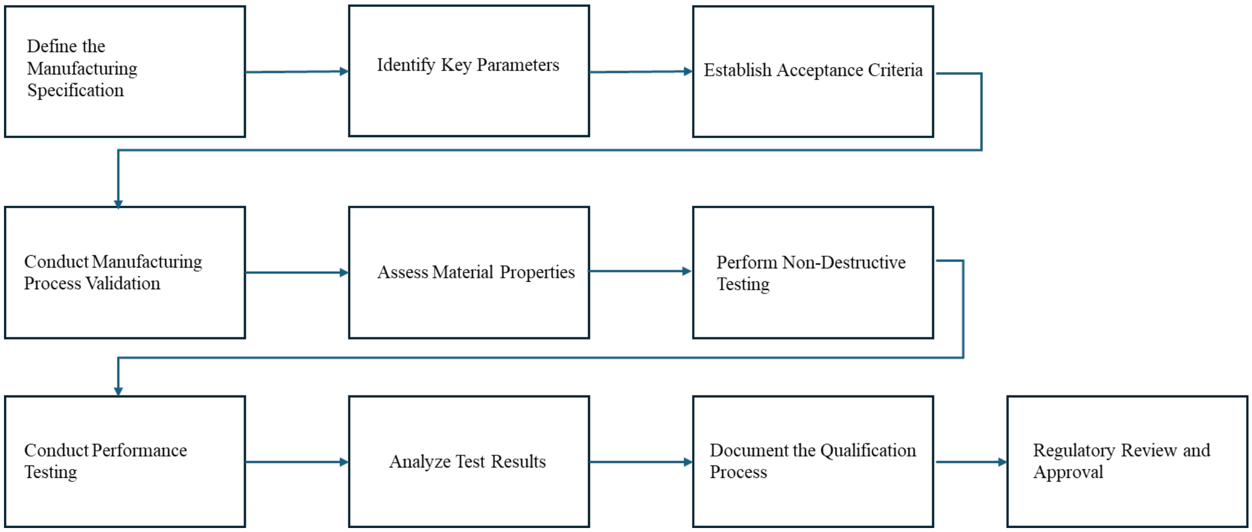
Table 4. Tests that can be used to quantify material properties.

Property	Recommended Tests
Thermal conductivity	Tests on unirradiated cladding samples over representative temperature range.
Thermal expansion	Tests on unirradiated cladding samples over representative temperature range.
Emissivity	None.
Enthalpy and specific heat	Tests on unirradiated cladding samples over representative temperature range.
Elastic modulus	Tensile tests on irradiated cladding tubes over a representative temperature range.

Property	Recommended Tests
Yield stress	Tensile tests on irradiated cladding tubes over a representative temperature range.
Thermal and irradiation creep	In-reactor creep tests on pressurized cladding tubes over a representative temperature range.
Axial irradiation growth	Poolside-length measurements from LRAs over a representative temperature range.
Corrosion rate	Poolside-length measurements from LRAs and thermal-hydraulic flow tests over representative range of burnups and temperatures.
Ballooning behavior	Burst tests, modified burst tests, and bend tests on unirradiated and irradiated cladding samples.
High-temperature corrosion	Corrosion tests on unirradiated and irradiated tube samples at relevant temperature.

The Phenomena Identification and Ranking Table (PIRT) process is used in the nuclear industry to systematically identify and rank the important phenomena affecting the behavior and performance of a specific system or component (NRC, 2020). The process involves assembling a multidisciplinary team of experts, identifying and categorizing the relevant phenomena, ranking the phenomena based on their importance, developing the EMs, validating the models, and using the results to inform analysis, design, and decision-making processes. The PIRT process helps ensure the most critical aspects are adequately addressed and that resources are focused on the key areas. It also facilitates collaboration and communication among experts, leading to a comprehensive understanding of the system or component under consideration. The PIRT process is iterative and can be revisited as new information becomes available or as the understanding of the system or component evolves. It is a useful tool in order to facilitate a discussion on the needs of the prototypic cladding product and then later in the process to capture any failure mechanisms that may have been missed or need further research.

Once the prototypic fuel-cladding product has been developed, licensing documentation should include sufficient information to ensure the control of key parameters affecting fuel performance during the manufacturing process, as shown and described in Figure 3. Fuel performance during normal operation and accident conditions can be highly sensitive to fuel manufacturing parameters. Manufacturing processes for a nuclear fuel product may evolve over the product life cycle; therefore, sufficient information should be provided in licensing documentation to provide reasonable assurance that cladding fabrication processes affecting fuel performance will be controlled during manufacturing.



- Define the Manufacturing Specification: Clearly define the manufacturing specification, including the requirements, processes, and controls for fabricating the fuel product.
- Identify Key Parameters: Identify the key parameters that significantly affect the behavior and performance of the fuel, such as material composition, dimensions, fabrication techniques, and quality control measures.
- Establish Acceptance Criteria: Establish acceptance criteria that define the desired fuel behavior and performance based on regulatory requirements, safety considerations, and operational needs.
- Conduct Manufacturing Process Validation: Validate the manufacturing process through experiments, tests, and inspections to ensure that the fuel product meets the defined manufacturing specification. This may involve prototype fabrication, process qualification tests, and quality control checks.
- Assess Material Properties: Assess the material properties of the fuel, such as mechanical strength, thermal conductivity, and chemical stability, to ensure they meet the specified requirements.
- Perform Non-Destructive Testing: Utilize non-destructive testing techniques, such as visual inspection, ultrasonic testing, and radiographic examination, to detect any defects or anomalies in the fuel product.
- Conduct Performance Testing: Conduct performance testing to evaluate the fuel's behavior under normal and abnormal operating conditions. This may involve thermal-hydraulic testing, irradiation testing, and mechanical testing.
- Analyze Test Results: Analyze the test results to determine if the fuel product meets the acceptance criteria defined in the manufacturing specification. Address and resolve any deviations or non-conformances.
- Document the Qualification Process: Document the entire qualification process, including the manufacturing specification, test procedures, test results, and any corrective actions taken. This documentation serves as evidence of compliance with the qualification requirements.
- Regulatory Review and Approval: Submit the qualification documentation to the regulatory authority for review and approval. The regulatory authority will assess the adequacy of the manufacturing specification and the supporting evidence to ensure compliance with regulatory requirements.

Figure 3. Cladding manufacturing qualification flowchart.

Additionally, key dimensions and tolerances for fuel clad components affecting performance should be specified. These dimensions and tolerances should be specific to components affecting fuel life-limiting failure and degradation mechanisms that are due to irradiation and exposure to the in-reactor environment (e.g., fuel pellet and cladding dimensions, key assembly dimensions).

The EM should be under development during the entirety of the cladding development process. Eventually, the basis of the EM and supporting data will need to be documented in a licensing topical report. The documentation should include sufficient information to ensure the control of key parameters affecting fuel performance and the data used to validate the model performance to the desired burnups.

To qualify an EM for assessing fuel performance, several steps can be taken. First, it is important to ensure the EM has the necessary capabilities to accurately model the geometry, material properties, and physical processes affecting fuel behavior. Any simplifications or assumptions made in the modeling should be justified. The EM should then undergo a validation process, which includes a peer review, calibration against the test data, and justification of the modeling scheme. It is crucial to assess the EM against the experimental data to establish uncertainties and biases in the model predictions. This assessment should cover the entire fuel performance envelope—including normal operation, AOOs, and postulated accident conditions. It is important to use experimental data that are independent from the data used to develop the EM and collect data over a test envelope that spans the fuel performance envelope. The data should be collected under an appropriate quality assurance program meeting regulatory requirements. Advanced modeling and simulation techniques can be used to inform the selection of constituents and systems and enable integral fuel performance analyses. Separate-effects tests can be conducted to obtain basic data on material behavior and property evolution under irradiation conditions. Physics-based models should be validated to support EM extrapolation beyond the limits of the available integral test data. Integral irradiation test data should be obtained to validate engineering scale fuel performance codes and confirm the performance and safety of the fuel system.

The following types of test data are typically needed:

- Experimental data on fuel behavior under various conditions—including normal operation, AOOs, and postulated accident conditions.
- Accurate material properties, such as thermal conductivity, mechanical properties, and fission-product release behavior.
- Information about the irradiation conditions—including neutron flux, temperature, and radiation dose.
- Detailed data on the fuel manufacturing process—including fabrication techniques, material composition, and quality control measures.
- Data on the boundary conditions, such as coolant flow rates, temperature profiles, and pressure conditions.
- Validation data from independent experiments to assess the accuracy and predictive capability of the model.
- Information about the uncertainties associated with the experimental data, material properties, and modeling assumptions.

Useful test data sources are presented in Table 5.

Table 5. Assessment data useful to validate fuel thermal-mechanical codes.

Assessment Data	Recommended Tests
Fuel centerline temperature Fission Gas Release	Instrumented irradiated integral tests at representative fuel burnups. LTA data and follow-up surveillance plan.
Rod internal pressure (RIP) and rod volume	LTA data and follow-up surveillance plan.
Cladding corrosion	Initially, none beyond the data in Table 4 and follow-up surveillance plan.
Cladding permanent hoop strain	Power ramp tests to assess the prediction of cladding strain following power ramp.

8. CONCLUSIONS

The SiC/SiC-composite claddings and fuel components are expected to provide excellent passive safety features both in design-basis accidents and design extension conditions severe accidents. This document outlined a pathway to qualifying new SiC/SiC-composite cladding highlighting the irradiation test data needs to satisfy regulatory requirements and develop appropriate SAFDLS. The introduction of multilayer SiC/SiC-composite cladding does not appear to warrant the establishment of new fuel-system damage criteria to qualify the new fuel-cladding type. The current criteria were found to be appropriate to develop SAFDLs for safe operation during normal operation, including AOOs. For a majority of the technical criteria, mechanical property and experimental data obtained from prototypic cladding designs operated under prototypical power histories are likely to be sufficient to verify compliance.

A few issues may be of concern for the multilayer SiC/SiC-composites. The chemical compatibility of SiC with the chemistry of the LWR coolant leading to a recession in the water is a concern.

Additionally, the low thermal conductivity and pseudo-ductile behavior of SiC will impact cladding mechanical response under pellet/clad mechanical interaction (PCMI) conditions. The 1% strain criterion will need to be modified to account for this behavior. Leak-tightness can be seen as a critical issue because it depends on the range of acceptable deformation, as well as the cladding design. Additionally, tests under DBA conditions, such as LOCA integral tests and RIA rod-ejection tests, must verify the behavior of the thermal/mechanical properties with the introduction of bend tests in place of the tube burst tests.

One item that should be investigated further with the goal of developing a new SAFDL is the potential for SiC/SiC-composite claddings to operate beyond DNB conditions. The current thermal-margin criterion is sufficient to demonstrate the avoidance of overheating from a deficient cooling mechanism. However, DNB is not necessarily a failure mechanism. Experiments have shown that SiC surfaces transfer heat better than the materials presently used for LWR fuel-rod cladding. SiC surfaces have also shown surface wettability and CHF results improved over stainless steel and Zircaloy surfaces. Mechanistic methods could be developed to demonstrate post-DNB survival and provide operating margin to the current thermal-margin criterion used for Zr-based cladding materials. Additionally, it may be necessary to prescribe a SAFDL or demonstrate successful end-plug joining technology.

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Appendix A

Technical Regulatory Criteria

The intent of this appendix is to discuss the current fuel damage criteria, as well as the steps taken by licensees to satisfy the criteria, and highlight any additional needs for regulatory acceptance criteria and/or testing requirements. This appendix follows the content found in Revision 3 of Section 4.2 of NUREG-0800 (NRC, 2007) as a guide to assure the fuel-system technical licensing/regulatory requirements are reviewed. Each design criteria are addressed in this appendix.

Fuel-System Damage

The design criteria presented here should not be exceeded during normal operation, including AOOs. The criteria discussed below assures the fuel-system dimensions remain within the operational tolerances and that functional capabilities are not reduced below those defined in the safety analysis.

Design Stress

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation or AOOs.

SRP Section 4.2: Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that are obtained by methods similar to those given in Section III of the American Society of Mechanical Engineers (ASME) Code are acceptable. Other proposed limits may need to be justified.

Technical Regulatory Requirement: Design stress represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - General Design Criteria (GDC) 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: In general, two methods are currently used to establish the design stress limit for fuel rods. The first employs the use of design ratios, defined as the ratio of effective stress to a stress limit. In this method, the stress limit is a conservative estimate of the unirradiated ultimate tensile stress. The design stress ratio approach is based on ANSI/ANS-57.5, "Light Water Reactor Fuel Assembly Mechanical Design and Evaluation." In the second method, the design stress limits are derived from the ASME Boiler and Pressure Vessel Code design criteria, as defined in Article III-2000 for Class 1 components. A total of four stress limits are used for fuel system structural components. These limits are based on the design stress intensity value (S_m), which is specified in Article III-2000 as the minimum of 2/3 the yield strength or 1/3 the ultimate tensile strength at either room or operating temperature. A detailed description of each limit is discussed in the following Design-Basis Approach section. These design stress limits ensure that stresses experienced by the fuel system structural components do not produce failure or gross distortion during normal operation and AOOs.

Design-Basis Approach: The fuel damage criteria for cladding stress ensure that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.

Cladding stress loads are calculated for conditions that bound normal operation and AOOs. Both finite element analysis methods and ASME stress analysis methods are employed to calculate the stress loads from the different mechanical contributions. The stress calculations are based on either reactor core beginning-of-life (BOL) or end-of-life (EOL) conditions and typical considerations include:

- Wall thickness variations to account for fabrication tolerances and corrosion

- Minimum and maximum RIP and coolant pressure variations
- Hot and cold coolant temperatures
- Locations in the cladding (e.g., inner surface vs. outer surface and mid-span vs. spacer grid).

The stress loads applied to fuel rods include hydrostatic pressure, spacer-grid contact, flow-induced vibrations, ovality, thermal and mechanical bow, and thermal gradient. In the design ratio approach, the stress loads from the different sources are combined using stress superposition and used to calculate the effective stress. The design stress ratio is calculated from the effective stress and ultimate tensile stress. Statistical methods are utilized to demonstrate the upper 95th percentile of the design ratio is less than unity.

For those methods based on Section III of the ASME Boiler and Pressure Vessel Code (BPVC), guidelines are provided to classify and combine the various stress loads. The classifications include Primary Membrane Stress (P_m), Primary Bending Stress (P_b), Primary Local Membrane Stress (P_L), and Secondary Stress (Q). Typical sources of cladding stress loads and classification are as follows:

- Hydrostatic pressure => primary membrane stress
- Grid spacer contact => primary local membrane stress
- Flow-induced vibrations => primary bending stress
- Ovality => primary bending stress
- Thermal and mechanical bow => secondary stress
- Thermal gradients => secondary stress.

Once the stress loads are calculated and combined to provide the different stress categories, the values are compared to the design stress intensity limits for each category, as defined below:

- P_m must not exceed S_m
- P_L must not exceed $1.5 \times S_m$
- P_m+P_b must not exceed $1.5 \times S_m$
- $P_L + P_b + Q$ must not exceed $3.0 \times S_m$.

New Baseline for Licensing

Assessment: The design stress limits used for fuel-rod cladding and other fuel system structural components appropriately address the effect of stress within the design process. However, new methods will need to be developed to account for the fact that in SiC cladding the major contribution of stress comes from the irradiation-induced swelling gradient across the thickness of the cladding and the internal pressure due to fission gas release. Mechanical-property data applicable to the target burnup level is required to verify the cladding yield stress at extended burnup remains larger than the value used to establish the design stress limit and does not display a decreasing trend with burnup. Corrosion data may be required to verify the wall thickness reduction used in the stress calculations accounts for the wall thickness reduction anticipated for all burnup and power levels.

Design Strain

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation or AOOs.

SRP Section 4.2: Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits obtained by methods similar to those given in Section III of the ASME Code are acceptable. Other proposed limits may need to be justified.

Technical Regulatory Requirement: Design strain represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: In general, two methods are used to establish the design strain limit for fuel rods. The first employs the use of design ratios, defined as the ratio of effective strain to a strain limit. In this method, the strain limit is a conservative estimate of the strain corresponding to the ultimate tensile strain. The design strain ratio approach is based on ANSI/ANS-57.5, "Light Water Reactor Fuel Assembly Mechanical Design and Evaluation." In the second method, the design strain limits are derived such that during normal operations the maximum uniform strain (elastic + plastic) due to uniform cladding creep and uniform cylindrical fuel pellet swelling and thermal expansion is less than a specified value based on mechanical-property characterization (typically 1%) from unirradiated conditions. In this context, uniform strain is defined as the circumferential average strain with a gauge length corresponding to the cladding dimensions.

For AOOs, the design limit for cladding strain is that the maximum uniform strain during the transient is less than a specified value based on mechanical-property characterizations (typically 1%). This value is sometimes reduced at burnups above 60 GWd/tU. The cladding design strain limits are set to ensure that strains experienced by the cladding do not produce cladding failure or gross distortion during normal operation and AOOs.

Design-Basis Approach: The fuel damage criteria for cladding strain ensure that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.

Cladding strains are calculated for conditions bounding normal operation and AOOs. Fuel performance codes are employed to calculate the strains from cladding creep and fuel pellet swelling and thermal expansion. The strain calculations are based on the worst-case loading conditions and reflect the cladding material conditions including wall thinning, the condition of the cladding oxidation layer including non-uniformities and local loss of oxide, cladding temperature, hydrogen content, and zirconium hydride distribution and orientation.

In the design strain ratio approach, statistical methods are utilized to demonstrate that the upper 95th percentile of the design strain ratio is less than unity.

New Baseline for Licensing

Assessment: The design strain limits used for fuel-rod cladding can appropriately address the pseudo-ductility effect of SiC/SiC-composite cladding within the design process. However, the 1% strain limit in the existing design limit is not appropriate for the introduction of SiC/SiC-composite cladding. A new design limit will need to be developed for SiC/SiC-composite cladding. This limit will need to be developed from mechanical-property data applicable to the target burnup level and the maximum material erosion including non-uniformities. Corrosion data may be required to verify that the wall thickness reduction used in the strain calculations represents the cladding condition anticipated for all burnup and power levels.

Design Fatigue

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation.

SRP Section 4.2: The cumulative number of strain fatigue cycles on the structural members mentioned in Sections B.2.1 and B.2.2 should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. Other proposed limits may need to be justified.

Technical Regulatory Requirement: Strain fatigue represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The design limit for strain fatigue is based on the cumulative fatigue usage factor for the cladding design stress and strain. Most licensees specify the cumulative (life) fatigue usage factor shall be less than 1.0, considering a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is more conservative. The O'Donnell and Langer design curve of stress amplitude versus number of cycles for irradiated Zircaloy is used in most cases (O'Donnel et al., 1964). This curve incorporates a safety factor of 2 on the stress amplitude and a safety factor of 20 on the number of cycles.

Design-Basis Approach: The fuel damage criteria for strain fatigue ensure that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The design approach employs the use of fuel performance codes to calculate the range of cladding strain (or stress) amplitudes during operation. A variety of methods are used to define the number of power cycles in the lifetime of a fuel rod, including load follow operation, AOOs, and selected DBAs. The calculations include the effects of fabrication tolerances on cladding thickness, internal pressure, fuel-cladding contact, etc.

New Baseline for Licensing:

Assessment: The O'Donnell and Langer design curve is not applicable for irradiated SiC/SiC-composite cladding. The design methods used to establish the strain fatigue fuel damage criteria will need to be defined and consider the impact of exposure on such parameters as pellet conductivity, pellet swelling, cladding erosion, and the increased residence time on the number of strain cycles.

Rod-to-Spacer-Grid Fretting

Baseline for Current Fuel-System Designs

Application: Fuel-system damage during normal operation.

SRP Section 4.2: Fretting wear at contact points on structural members should be limited. The allowable fretting wear should be stated in the Safety Analysis Report (SAR) and the stress and fatigue limits in the design stress and design strain. Design fatigue should presume the existence of this wear.

Technical Regulatory Requirement: Fretting wear represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The design limit for fretting wear is that fuel-rod failures due to fretting shall not occur.

Design-Basis Approach: The design limit for fretting wear ensures that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The design-basis approach employs the use of data taken from operating reactors (experience) and out-of-reactor flow tests to provide assurance that the design limit is met for both Zircaloy and Inconel grid designs.

New Baseline for Licensing

Assessment: LTAs applicable to the target burnup level may need to be examined to verify that widespread fuel-rod-to-grid fretting wear is not a problem or could lead to fuel-rod failure. Cladding failure by fuel-rod-to-grid fretting occurs rapidly following the development of high amplitude flow-induced vibrations. The addition of the hard monolithic SiC may also wear grid features which may lead to failure. Flow anomalies are generally the cause of these vibrations and are more related to the position of the assembly within the core than fuel assembly burnup.

Compared to metals, ceramics show a much wider statistical variation in failure stresses and strains. It may be necessary to account for this by generated a larger failure database to develop a statistical distribution for failure probability calculations.

Design Oxidation

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation or AOOs.

SRP Section 4.2: Oxidation should be limited. Allowable oxidation levels should be discussed in the SAR and shown to be acceptable.

Technical Regulatory Requirement: Oxidation of primary RCS surfaces tends to form corrosion products (CRUD) on the fuel-cladding outer surface. A limit on CRUD represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The design limit for oxidation is that the spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members shall not be damaged due to excessive oxidation. Design methods do not uniformly specify a limit for maximum oxide thickness, although most PWR vendors apply a 100 μm limit. The effects of the outer surface (and inner surface where appropriate) oxidation are included in thermal and mechanical analyses of the fuel system.

Design-Basis Approach: The consideration of the effects of oxidation on the cladding, grid spacer surfaces, and other fuel system structural components ensures that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The SRP Section 4.2 guidelines do not require an explicit limit on the maximum allowable oxide thickness; however, the guidelines suggest that the impact of oxidation on the thermal and mechanical performance should be considered in the design analysis when comparing to the design stress and strain limits. Wall thinning caused by oxidation increases the stress on the fuel system structural members and the higher temperature decreases the material strength. These effects may need to be considered in the design analyses to ensure the cladding does not exceed the mechanical design limits (e.g., design stress and design strain). The oxidation thickness used in the design analysis is based on experimental data obtained from poolside examinations.

New Baseline for Licensing

Assessment: SiC/SiC-composite cladding is susceptible to erosion wear in a PWR coolant environment. This erosion of the monolith layer during operation can lead to wall thinning. Experimental data obtained from fuel rods operated under prototypical power histories are required to verify that material erosion considerations in the thermal and mechanical design methods are appropriate.

Hydriding

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation or AOOs.

SRP Section 4.2: Hydriding should be limited. Allowable hydriding levels should be discussed in the SAR and shown to be acceptable. These levels should be presumed to exist in Sections B.2.1, B.2.2, and B.2.3, respectively.

Technical Regulatory Requirement: A limit on hydriding represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The design limit for hydriding is that the spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members shall not be damaged due to excessive hydriding. Design methods either specify a limit for the amount of hydrogen pickup in the cladding such that the mechanical properties of the cladding material are not adversely affected or reflect hydrogen effects in the mechanical-property formulations.

Design-Basis Approach: The consideration of the effects of hydrides on the fuel system structural member mechanical properties ensures that cladding functional capabilities are not reduced below those assumed in the safety analysis. The SRP Section 4.2 guidelines do not require an explicit limit on the maximum allowable hydrogen pickup in the fuel system structural member; however, the guidelines suggest that the impact of hydrides on the mechanical performance should be considered in the design analysis when comparing to the design stress and strain limits. Cladding hydrogen pick up limits are required to prevent excessive degradation of mechanical properties due to hydrogen embrittlement by the formation of zirconium hydride platelets when hydrogen is released during the oxidation process. This effect may need to be considered in the design analyses to ensure that the fuel system structural member does not exceed the mechanical design limits (e.g., design stress and design strain). The hydrogen pickup rate used in the design analysis is based on experimental data obtained from prototypic experiments.

New Baseline for Licensing

Assessment: The phenomena is not applicable to SiC/SiC-composite cladding.

CRUD

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation.

SRP Section 4.2: Buildup of corrosion products (CRUD) should be limited. Allowable CRUD levels should be discussed in the SAR and shown to be acceptable. These levels should be presumed to exist in Sections B.2.1, B.2.2, and B.2.3, respectively. The effect of CRUD on thermal-hydraulic considerations is reviewed as described in SRP Section 4.4.

Technical Regulatory Requirement: A limit on CRUD represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The design limit for CRUD is that the fuel-rod system shall not be damaged due to excessive CRUD buildup. Most design methods do not specify a limit for the CRUD layer. The effects of the CRUD layer are included in thermal analysis of the fuel rod.

Design-Basis Approach: The consideration of the effects of CRUD deposition on the cladding and grid spacer surfaces ensures that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The SRP Section 4.2 guidelines do not require an explicit limit on the maximum allowable CRUD layer; however, the guidelines suggest that the impact of CRUD on the thermal and mechanical performance should be considered in the design analysis when comparing to the design stress and strain limits. Bounding values of CRUD are used in the thermal analysis to calculate temperatures. Wall thinning caused by oxidation increases the cladding stress and the higher temperature decreases the material strength. These effects may need to be considered in the design analyses to ensure that the cladding does not exceed the mechanical design limits (e.g., design stress and design strain). The CRUD layer used in the design analysis is based on experimental data obtained from poolside examinations.

A second impact that may need to be considered is the reduction in assembly flow area and increase in surface roughness with CRUD deposition. Consideration of such CRUD effects are normally included in the applicable thermal-hydraulic evaluations.

New Baseline for Licensing

Assessment: Experimental data obtained from fuel rods operated under prototypical power histories may be required to verify the CRUD layer considerations in the thermal and mechanical design methods are appropriate. In addition, the potential for AOO consequences in PWRs also may need to be addressed.

Rod Bow

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation.

SRP Section 4.2: Dimensional changes such as rod bowing need not be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

Technical Regulatory Requirement: The effects of rod bow represent a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: Most PWR licensees utilize a DNBR penalty that is a function of rod bow (channel closure) and a local power peaking penalty, rather than placing design limits on the amount of rod or assembly bow.

For BWRs, channel bow also has an impact in the bundle local power peaking factors and may imply a penalty on the MCPR calculation similar to the impact of rod or assembly bow on the DNBR calculations for a PWR. Some BWR licensees apply a penalty to the bundle R factor (a parameter related to the local peaking factors) to take into consideration the calculated channel bows of the bundles in the core rather than placing design limits on the amount of channel bow.

Design-Basis Approach: Incorporating the effects of rod, assembly, or channel bow on the design analysis ensures that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.

The design-basis approach for licensees with PWRs utilizes a rod bow correlation developed from rod-to-rod gap measurements obtained from post-irradiation examinations. This correlation provides the magnitude of rod bow as a function of fuel assembly burnup, operating conditions, etc. The 95% upper bound of the rod bow data is used in the calculation of the DNBR penalty factor. Out-of-pile CHF test data are used to develop a DNBR correlation as a function of rod-to-rod gap distance. Combining the rod bow correlation with the DNBR correlation provides a DNBR penalty as a function of fuel assembly burnup. Power peaking uncertainties due to rod or assembly bow are included in the calculation of the power operating limits (peaking factors). The local power peaking uncertainties account for local neutron moderation variations caused by rod or assembly bow.

New Baseline for Licensing

Assessment: Fuel rod, assembly, and channel bow data applicable to the target burnup level may be required to verify that the effects of rod, assembly, and channel bow on DNBR/MCPR and power peaking are not underestimated.

Irradiation Growth

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation.

SRP Section 4.2: Dimensional changes such as irradiation growth of fuel rods, control rods, and guide tubes need not be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

Technical Regulatory Requirement: A limit on fuel rod and assembly irradiation growth represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: For some BWR designs, the fuel-rod irradiation growth limit is to design sufficient clearance between the fuel rod and the upper tie plate so that axial growth will not cause the fuel-rod expansion spring to go solid (e.g., full closure).

The PWR assembly irradiation growth design limit is to maintain sufficient clearance between the fuel assembly upper nozzle and the upper core plate so that irradiation growth of the assembly will not cause the assembly hold-down spring to go solid (e.g., full closure).

Design-Basis Approach: The initial fuel-rod-to-nozzle gap is designed to allow for differential growth between the fuel rod and the fuel assembly guide tubes, as founds in PWRs, or the tie rods, as found in BWRs, to preclude interference during the fuel design lifetime. To accomplish this, a conservative assessment of the differential fuel rod and assembly growth is performed, including fabrication tolerances. This method is used to calculate the change in the fuel-to-nozzle (or tie plate) gap as a function of irradiation. Post-irradiation examination data is used to develop the fuel-rod growth, fuel assembly growth, and gap closure models as a function of fast neutron fluence or exposure.

The initial fuel assembly-to-upper core-plate gap in PWRs is designed to accommodate fuel assembly growth without fully collapsing the assembly hold-down spring. This gap is designed assuming the maximum assembly growth and worst-case fabrication tolerances and an upper bound (typically 95/95) of the fuel-rod growth is combined with the lower bound assembly growth and worst-case fabrication tolerances. For BWRs, the fuel assembly is designed to preclude both full compression of the expansion spring and disengagement of individual fuel rods from the upper tie plate due to fuel rod and assembly differential growth.

New Baseline for Licensing

Assessment: Axial growth is an important design consideration for the reliable operation of a fuel assembly. In addition, differential swelling across the cladding cross-section can also cause rod bowing. Irradiation growth of the fuel rods sufficient enough to cause interference between the fuel rod ends and the upper nozzle (tie plate) may cause rod bow. Fuel-rod bow may result in rod-to-rod contact and failure of the cladding. In addition, fuel assembly irradiation growth may cause collapse of the fuel assembly hold-down spring in PWRs, which could cause fuel assembly bow. Fuel assembly bow may inhibit the insertion of the control rods.

Fuel surveillance programs generally measure the fuel-rod length and the overall fuel assembly length. This data verifies that the fuel-rod-to-nozzle gap decreases with increasing fluence.

For the introduction of SiC/SiC-composite cladding, data may be required to demonstrate that the fuel rod and assembly irradiation growth models used in the fuel assembly design process are appropriate. This may require information on anticipated fast fluence levels at discharge. The fuel assembly design process may need to balance the fuel rod and fuel assembly growth to assure that appropriate component clearances/engagements are maintained.

Internal Gas Pressure

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation.

SRP Section 4.2: Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.

Technical Regulatory Requirement: RIP represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The RIP limit used for licensing current fuel-rod designs is greater than the nominal coolant pressure. For modern fuel designs, the RIP limit has been established at a level below that required to cause: (1) fuel-cladding gap reopening due to cladding creep-out during constant and increasing fuel-rod power conditions under normal operation, when the RIP exceeds the system pressure, and (2) extensive DNB propagation to occur. Thermal and mechanical calculations have been performed to provide the engineering justification for using a RIP limit exceeding RCS pressure.

Design-Basis Approach: The RIP fuel damage criteria ensures that fuel-system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The typical design approach employs the use of fuel performance computer codes that calculate the evolution of the RIP as a function of the fuel design, burnup, and operating conditions to evaluate the possibility of gap reopening combined with the thermal-hydraulics codes to ensure that extensive DNB propagation does not occur. In general, the methods include the effects of initial helium pressure/inventory, initial internal void volume, fission gas release, helium release from the fuel pellet and burnable poisons, total rod internal void volume, and the temperature of the various internal void volumes. The input power history is developed using NRC-approved methods and may include several limiting transients to add conservatism to the analysis. Some approaches may apply conservatisms on best-estimate models to ensure bounding calculations for maximum RIP.

New Baseline for Licensing

Assessment: The validity of the design-basis calculations for RIP needs to be verified using applicable data for the burnup range of interest. This includes comparison of analysis results to fission gas release or RIP measurements for the discharge burnup of interest.

Hydraulic Lift Loads

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation, AOOs, and accidents.

SRP Section 4.2: Worst-case hydraulic loads for normal operation should not exceed the hold-down capability of the fuel assembly either from gravity or from the hold-down springs. Hydraulic loads for this evaluation are reviewed as described in SRP Section 4.4.

Technical Regulatory Requirement: A limit on hydraulic uplift loads represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The design limit for the hydraulic uplift load is that the fuel assembly hold-down capability (e.g., wet weight and/or hold-down forces) should maintain engagement of the fuel assembly with the lower support plate during normal operation and accident events.

Design-Basis Approach: The design bases for the fuel assembly hold-down capability considers the hydraulic uplift loads, which are a function of the coolant flow and the assembly pressure drop, as well as the hold-down forces, such as the assembly weight and upper nozzle spring. For PWRs, the design approach includes the effects of irradiation-induced relaxation of the spring and assembly growth on the upper nozzle spring forces. Hydraulic uplift loads are identified for a variety of flow and pressure conditions depending on reactor design.

New Baseline for Licensing:

Assessment: The fuel design limit for the hydraulic uplift loads is to maintain engagement of the assembly with the lower core support plate under normal operation and accident conditions. This limit is independent of the introduction of SiC/SiC-composite cladding. The analysis methods used to design the assembly hold-down spring in a PWR should consider the effects of extended burnup operation and/or increased enrichment on the spring design. These effects include irradiation-induced relaxation of the spring material and the fuel assembly irradiation growth.

If the spring force relaxation with fast neutron fluence is less than anticipated or complete closure of the spring occurs, it is possible that the spring hold-down force can increase with irradiation due to assembly growth, causing higher axial loads to be imposed on the assembly. Depending on the design of the guide tubes, high axial loads on the assembly may produce lateral deflections of the guide tubes sufficient to impede the insertion of the control rod assembly during shutdown. Therefore, the design process of the assembly hold-down spring may need to consider the assembly axial load capacity to ensure that lateral deflections are minimized during extended burnup operation.

The data typically used to verify compliance includes fuel assembly growth data, channel dimensional performance data, hold-down spring stiffness and force data, and fuel assembly lateral deflection data. This data may need to be obtained from fuel assemblies irradiated under prototypical power history conditions.

Control Rod Reactivity and Insertability

Baseline for Current Fuel-System Designs:

Application: Fuel-system damage during normal operation and AOOs.

SRP Section 4.2: Dimensional changes resulting in PWR fuel assembly lateral deflection (bow) or BWR channel lateral deflection (bow) during normal operation should not result in incomplete insertion of mechanical reactivity control devices.

Technical Regulatory Requirement: A limit on the amount of PWR fuel assembly bow or BWR channel bow represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: PWR fuel assembly bow and BWR channel bow must be accounted for such that mechanical reactivity control device insertability is maintained.

Design-Basis Approach: In the past, to ensure that control rod insertability is maintained and the effects of fuel assembly bow accounted for, rod drop tests were performed periodically with increasing core average burnup. In addition, limits are sometimes placed on fuel assembly burnups under control positions in the core. Also, to limit the amount of lateral deformation in fuel assemblies, some vendors have stiffened the guide tubes.

New Baseline for Licensing

Assessment: Design limits used for assembly and channel bow can address the introduction of SiC/SiC-composite cladding within the design process. The PWR fuel assembly bow and BWR channel bow data applicable to the target burnup level may be required to verify that the effects of assembly and channel bow are not underestimated and therefore do not inhibit control rod or control blade insertability.

Fuel-Rod Failure

The design criteria presented in this section applies to normal operation, AOOs, and DBAs. When the failure thresholds are applied to normal operations including AOOs, they are used as SAFDLs since fuel failure under those conditions should not occur according to the traditional interpretation of 10 CFR 50, Appendix A, GDC 10. When these thresholds are used for DBAs, fuel failures are allowed but may need to be accounted for in the dose calculations required by 10 CFR 50.67 and 10 CFR 100.

Internal Hydriding

Baseline for Current Fuel-System Designs:

Application: Fuel-rod failure during normal operation.

SRP Section 4.2: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities within the fuel very low during fabrication. Acceptable moisture levels for Zr-based alloy-clad UO_2 fuel should be no greater than 20 µg/g (20 ppm). Current American Society for Testing and Materials (ASTM) specifications for UO_2 fuel pellets state an equivalent limit of 2 µg/g (2 ppm) of hydrogen from all sources. For other materials clad in Zr-based alloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 mg H_2O per cm^3 of hot void volume within the Zr-based alloy cladding has been shown to be insufficient for primary hydride formation.

Technical Regulatory Requirement: A limit requiring no cladding failure by internal hydriding represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The design limit for internal hydriding is that no fuel failures will occur by internal (primary) hydriding. Most design methods use a maximum allowable level of moisture (or hydrogen) within the fuel pellets to ensure compliance to the design limit.

Design-Basis Approach: The design-basis approach is to use manufacturing techniques to minimize the moisture content within the fuel pellets.

New Baseline for Licensing

Assessment: Hydriding of SiC/SiC-composites is not an issue. However, metal liners/liquid metals in the fuel-cladding gap used to improve gap thermal conductivity may be affected by hydriding. Advances in manufacturing techniques for high-density fuel pellets (> 95% TD) have effectively eliminated internal hydrogen sources within the cladding, thus precluding the hydriding sources.

Cladding Collapse

Baseline for Current Fuel-System Designs:

Application: Fuel-rod failure during normal operation.

SRP Section 4.2: If axial gaps in the fuel pellet column occur due to densification, then the cladding has the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.

Technical Regulatory Requirement: A limit requiring no cladding failure by cladding creep collapse represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and

35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

Design Limit: The design limit for cladding collapse is that no fuel-rod failures shall occur by flattening of the cladding.

Design-Basis Approach: The design-basis approach includes the use of NRC-approved cladding collapse models to show that the in-reactor residence times associated with the target rod average burnup level will not result in cladding collapse. These models include the effects of such parameters as pellet densification, cladding thickness, cladding ovality, the mechanical properties of the Zr-based alloy cladding, cladding temperature, RIP, coolant pressure, and pellet-cladding contact. Either a collapsed rod frequency of unity or a maximum ovality is used as the design limit for the analysis models. Conservative assumptions are used in the analysis methods for the amount fuel densification (size of the axial gaps), the initial RIP and pellet-cladding interaction. These assumptions bound the expected performance of the fuel column and cladding.

New Baseline for Licensing

Assessment: Current methods to calculate the occurrence of clad flattening are not appropriate for SiC/SiC-composite cladding. The inherent strength of the composite material precludes the effects of clad flattening for SiC/SiC-composite cladding. Therefore, cladding collapse is not an appropriate criterion nor necessary for the introduction of SiC/SiC-composite cladding.

Overheating of Cladding

Baseline for Current Fuel-System Designs:

Application: Fuel-rod failure during normal operation, AOOs, and postulated accidents.

SRP Section 4.2: It has been traditional practice to assume that failures will not occur if the thermal-margin criteria (e.g., DNBR for PWRs and MCPR for BWRs) are satisfied. The review of these criteria is detailed in SRP Section 4.4. For normal operation and AOOs, the violation of thermal-margin criteria is not permitted. For DBAs, the total number of fuel rods that exceed the criteria are assumed to fail for radiological dose calculation purposes.

Although a thermal-margin criterion is sufficient to demonstrate the avoidance of overheating from a deficient cooling mechanism, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. There is at present little experience with other approaches, but new positions recommending different criteria may need to address cladding temperature, pressure, time, duration, and oxidation.

Technical Regulatory Requirement: A limit requiring no fuel-rod failure by exceeding the thermal-margin criteria during normal operation or AOOs represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

For postulated DBAs, fuel-rod failure is permitted. However, 10 CFR 50.67 and 10 CFR 100 require that the fission-product release from rods that exceed the thermal-margin criteria be limited to meet offsite radioactivity dose limits.

Design Limit: The design limit for overheating of the cladding is that the thermal-margin criteria will not be exceeded during normal operation and AOOs. For DBAs, the number of fuel rods exceeding the thermal-margin criteria are summed to include in the fission-product release analysis.

Design-Basis Approach: The design-basis methods utilize thermal-hydraulic analysis codes to calculate the maximum clad temperature during normal operation, AOOs, and DBAs.

The thermal limits for both normal operation and transient conditions are derived such that appropriate thermal margin is maintained by selecting an MCPR based on a statistical analysis such that 99.9% of the fuel rods would be expected to avoid boiling transition. Similarly, the DNBR limits for both normal operation and transient conditions are derived such that there is a 95% probability at a 95% confidence level that DNB will not occur on a fuel rod.

New Baseline for Licensing:

Assessment: The requirement that the thermal-margin criteria may not be exceeded during normal operation and AOOs is applicable for SiC/SiC-composite cladding. However, a time at temperature approach in place of DNB limit would leverage the improved thermal-hydraulic characteristics of SiC/SiC cladding providing margin to the current methods used.

Overheating of Fuel Pellets

Baseline for Current Fuel-System Designs:

Application: Fuel-rod failure during normal operation, AOOs, and DBAs.

SRP Section 4.2: It has been traditional practice to assume that failure will occur if centerline melting takes place. This analysis should be performed for the maximum LHGR anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and AOO, centerline melting is not permitted. For DBAs, the total number of rods experiencing centerline melting should be assumed to fail for radiological dose calculation purposes. The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to come into contact with the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.

Technical Regulatory Requirement: A limit requiring no fuel centerline melting during normal operation or AOOs represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

For DBAs, fuel centerline melt is permitted. However, 10 CFR 50.67 and 10 CFR 100 requires that the fission-product release from rods experiencing fuel melting be limited to meet the offsite radioactivity dose limits.

Design Limit: The design limit for overheating of the fuel pellets is that the fuel melting temperature will not be exceeded during normal operation and AOOs. For DBAs, the number of fuel rods exceeding the fuel pellet melting temperature are summed to include in the fission-product release analysis.

Design-Basis Approach: The design-basis methods utilize fuel performance analysis codes to calculate the maximum fuel temperature during normal operation, AOOs, and postulated DBA conditions. Bounding power histories or the use of actual rod power histories along with a statistical treatment of other analysis inputs are used such that the predicted fuel temperature represents an upper bound. The effects of burnup on the fuel thermal conductivity and the fuel melting temperature are included through either modified material models or penalty factors.

New Baseline for Licensing

Assessment: For normal operation and AOOs, the maximum achievable fuel temperature is limited to ensure conformance with the design and licensing limits, including the fuel temperature limits. The analysis methods used to calculate the fuel temperature may need to include the effects of enrichment on assumed bounding power and burnup on the heat conduction through the fuel rod, such as the fuel thermal conductivity, the fuel-cladding gap conductance, and the outer cladding surface CRUD layer formation. This is very important for DBAs where fuel temperatures may exceed the power level restrictions for normal operation. Also, the maximum temperature limit employed in the design-basis analysis of DBAs may need to be verified to not exceed the fuel melting temperature at target burnup.

Excessive Fuel Enthalpy

Baseline for Current Fuel-System Designs:

Application: Fuel-rod failure during postulated RIAs, including control rod-ejection accident (CREA) or control rod drop accident (CRDA).

SRP Section 4.2: The sudden increase in fuel enthalpy from an RIA below fuel melting can result in fuel failure due to pellet/clad mechanical interaction (PCMI) (see SRP Section 4.2, Subsection II, item 1.B.vii). Exceeding the DNBR for a PWR or the critical power ratio (CPR) for a BWR may result in cladding failure during an RIA.

Technical Regulatory Requirement: RIAs consist of DBAs, which involve a sudden and rapid insertion of positive reactivity. These accident scenarios include a CREA for PWRs and a CRDA for BWRs. The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion that promptly increases local core power. Fuel temperatures rapidly increase, prompting fuel pellet thermal expansion. The reactivity excursion is initially mitigated by Doppler feedback and delayed neutron effects followed by reactor trip. NUREG-0800 Sections 15.4.8 and 15.4.9 provide further detail on the CREA and CRDA, respectively (NRC, 2007).

The total number of fuel rods that may need to be considered in the radiological assessment is equal to the sum of all fuel rods failing due to each of the following criteria. Applicants do not need to double-count fuel rods that are predicted to fail for more than one of the criteria.

1. The high-cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure. For intermediate (greater than 5% rated thermal power) and full power conditions, fuel-cladding failure is presumed if local heat flux exceeds the thermal design limits (e.g., DNBR and CPR).
2. The PCMI failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Appendix B of SRP Section 4.2.

Fuel-cladding failure may occur almost instantaneously during the prompt fuel enthalpy rise due to PCMI or may occur as total fuel enthalpy, which considers both prompt and delayed components, heat flux, and cladding temperature increase. For the purpose of calculating fuel enthalpy for assessing PCMI failures, the prompt fuel enthalpy rise is defined as the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt pulse. For assessing high-cladding temperature failures, the total radial average fuel enthalpy may need to be used.

Fuel-rod thermal-mechanical calculations, employed to demonstrate compliance with Criteria 1 and Criteria 2 below, may need to be based upon design-specific information accounting for manufacturing tolerances and modeling uncertainties using NRC-approved methods including burnup-enhanced effects on pellet power distribution, fuel thermal conductivity, and fuel melting temperature.

1. Peak radial average fuel enthalpy must remain below 230 cal/g.
2. Peak fuel temperature must remain below incipient fuel melting conditions.
3. Mechanical energy generated as a result of: (1) non-molten fuel-to-coolant interaction, and (2) fuel-rod burst, must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
4. No loss of coolable geometry due to: (1) fuel pellet and cladding fragmentation and dispersal, and (2) fuel-rod ballooning.

Total fission-product gap fraction available for release following any RIA would include the steady-state gap inventory present prior to the event, plus any fission gas released during the event. The steady-state gap inventory would be consistent with the non-LOCA gap fractions cited in NRC Regulatory Guide

(RG) 1.183 (Table 3) and RG 1.195 (Table 2) and would be dependent on operating power history. Whereas fission gas release (FGR) into the rod plenum during normal operation is governed by diffusion, pellet fracturing and grain boundary separation are the primary mechanisms for FGR during the transient. The combined steady-state gap inventory and transient FGR from every fuel rod predicted to experience cladding failure (e.g., all failure mechanisms) may need to be used in the dose assessment.

General Design Criterion 28 (GDC 28) defined in 10 CFR Part 50 Appendix A specifies that reactivity control systems should be designed to assure the effects of a postulated reactivity accident neither: (1) result in damage to the RCPB greater than limited local yielding, or (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to cause serious impairment of core cooling capability. These limits are related to pressure vessel integrity and core coolability and are not concerned with cladding failure (e.g., loss of fuel-rod hermeticity).

Design Limit: Because PWR rod ejection and BWR rod drop accidents are classified as Condition IV events, fuel-rod failure may occur during these events—provided offsite and control room radiological consequences remain within acceptable limits. Specific guidance on the implementation of GDC 28 requirements have been described in RG 1.236 and the SRP. In particular, RG 1.236 identifies acceptable analytical methods and assumptions, as well as the following acceptance criteria to address GDC 28:

1. Peak radial average fuel enthalpy limited to 230 cal/g.
2. Maximum RCS pressure limited to the value that will cause stresses to exceed an Emergency Condition (Service Level C), as defined in Section III of the ASME BPVC.
3. Offsite dose consequences limited to "well within" the guidelines in 10 CFR 50.67 and 10 CFR 100. The basis of the first criteria on fuel enthalpy is to maintain a coolable geometry. As discussed in RG 1.236, the current RG 1.77 criterion of 280 cal/g has been judged inappropriate to ensure fuel-rod geometry and long-term coolability are maintained.

Design-Basis Approach: The approach to demonstrate compliance to the excessive fuel enthalpy limit for an RIA consists of the following steps:

- Calculate the energy deposition following a rapid insertion of reactivity using conservative neutron kinetics methods and assumptions
- Perform a thermal analysis using the energy deposition to identify the maximum radially averaged fuel enthalpy.

The methods and assumptions used to perform the neutron kinetics calculations are dependent on the specific methodology, fuel design, and fuel management scheme. Areas requiring conservative considerations include:

- Ejected rod worth
- Reactivity-insertion rate
- Delayed neutron fraction and prompt neutron lifetime
- Coolant pressure, inlet temperature, and mass flow rate
- Fuel-rod heat transfer properties
- Moderator temperature and Doppler coefficients
- Reactivity insertion during trip versus time and reactor trip delay time.

The results of static core depletion calculations are used to initialize the nuclear parameters and define the fuel assembly peaking factors for worst-case conditions, typically at the end of cycle. Based on the nuclear parameters listed above, either point, 1D, 1.5D or 3D spatial kinetics methods are used to calculate the core average power pulse transient for the ejection/drop of the highest worth control rod.

The hot spot power transient is calculated from the assembly peaking factors based on the limiting xenon distribution during the entire cycle. This information is input into the thermal heat conduction analysis to calculate the radially average fuel pellet enthalpy. Normally, adiabatic pellet heat-up is assumed in the calculation of the fuel enthalpy to obtain maximum values.

New Baseline for Licensing:

Assessment: The analysis methods used to calculate the excessive fuel enthalpy may have to include the effects of enrichment and burnup on the fuel reactivity and cladding material properties. The methods currently used are appropriate to verify compliance with the excessive fuel enthalpy limit.

Pellet/Cladding Interaction

Baseline for Current Fuel-System Designs:

Application: Fuel rod failure during normal operation, AOOs, and DBAs.

SRP Section 4.2: There is no criterion that currently exists for fuel failure resulting from PCI or PCMI. PCI is generally caused by stress-corrosion cracking due to fission-product (iodine) embrittlement of the cladding, while PCMI is primarily a stress-driven failure.

Two related criteria should be applied, but they are not sufficient to preclude PCI/PCMI failures. First, the uniform strain of the cladding should not exceed 1%. In this context, elastic and inelastic uniform strain is defined as transient-induced deformation with gauge lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Mechanical testing may need to demonstrate the irradiated cladding ductility at maximum waterside corrosion (hydride embrittlement) is well within the 1% strain criterion. Although observing this strain limit may preclude some PCI/PCMI failures, it may not preclude the corrosion-assisted failures occurring at low strains, nor may it preclude highly localized overstrain failures. Second, fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding.

Technical Regulatory Requirement: As noted above, there are no specifically applicable criteria for PCI failures. However, the NRC uses two acceptance criteria to help prevent PCI/PCMI failures: (1) less than 1% transient-induced elastic and inelastic cladding strain, and (2) no centerline fuel melting.

The NRC does expect that analyses of cladding strain for normal operation, AOOs, and DBAs apply approved fuel thermal expansion and gaseous fuel swelling models, as well as irradiated cladding properties.

For DBAs, fuel-rod failure is permitted. However, 10 CFR 100 requires that the fission-product release from rods exceeding the thermal-margin criteria be limited to meet offsite radioactivity dose limits

Design Limit: The design limit for the transient-induced uniform strain of the cladding should not exceed 1% and centerline melt should be avoided. Centerline melt is assumed to occur when the fuel temperature reaches the melting point.

Design-Basis Approach: The design-basis methods utilize fuel performance analysis codes to calculate the maximum fuel temperature during normal operation conditions, AOOs, and DBAs. Bounding power histories and statistical methods are used in the analysis approach to ensure the highest fuel temperatures are calculated. The effects of burnup and integral burnable absorber material on the fuel pellet thermal conductivity and fuel pellet melting temperature are included either through modified material models or penalty factors.

Cladding strains are calculated for conditions bounding the expected conditions. Fuel performance codes are employed to calculate the strains from cladding creep and fuel pellet swelling and thermal expansion. The strain calculations are based on the worst-case loading conditions and cladding geometry, including wall thinning. These codes must be approved by the NRC.

To avoid the occurrence of PCI failures, BWR vendors have at times introduced operating restriction guidelines. Similarly, PWR vendors recommend start-up ramp rate limits based on ramp rate test data. These restrictions have reduced the incidence of PCI failures and complement the 1% criterion.

New Baseline for Licensing:

Assessment: The analysis methods used to calculate the fuel temperature may need to include heat conduction through the fuel rod such as the fuel thermal conductivity, the fuel-cladding gap conductance, and outer CRUD layer formation.

A new transient uniform strain limit used for PCI/PCMI will need developed for SiC/SiC-composite cladding since microcracking can occur at strains as lower than current cladding types. Operational limits or other means may be needed to reduce the strains created during power ramps. Mechanical-property data applicable to the target burnup level, the maximum thickness erosion, and CRUD effects may be required to verify the cladding has sufficient ductility at extended burnup to satisfy the design strain limit for both normal operations, AOOs, and DBAs. Corrosion data may be required to verify the wall thickness reduction.

Bursting

Baseline for Current Fuel-System Designs:

Application: Fuel-rod failure during normal operation, AOOs, and DBAs.

SRP Section 4.2: To meet the requirements of 10 CFR 50.46 as the requirements relate to ECCS performance evaluation, a calculation of the swelling and rupture (i.e., bursting) of the cladding resulting from the temperature distribution in the cladding and from pressure differences between the inside and outside of the cladding should be included in the ECCS EM. RG1.157 provides guidelines for performing a realistic (i.e., best-estimate) model to calculate the degree of cladding swelling and rupture. Alternatively, Appendix K of 10 CFR 50 presents acceptable features of an EM for predicting the degree of swelling and rupture in Zr-based alloy cladding. Although fuel suppliers may use different rupture-temperature vs. differential-pressure curves, an acceptable curve should be similar to the one described in NUREG-0630, “Cladding Swelling and Rupture Models for LOCA Analysis,” in April 1980.

Technical Regulatory Requirement: There are no specifically applicable criteria associated with cladding rupture other than the 10 CFR 50 requirement that the degree of swelling not be underestimated, and the balloon not block the coolant channel.

Design Limit: Fuel cladding will balloon under certain combinations of temperature, heating rate, and stress during a LOCA. There are no specific design limits associated with cladding rupture other than the degree of swelling is not underestimated and the balloon is not allowed to block the coolant channel.

Design-Basis Approach: The design-basis methods utilize cladding deformation and rupture models directly coupled to models for cladding ballooning and flow blockage used in NRC-approved LOCA-ECCS EMs. All parameters that are important to LOCA analysis may need to be appropriately accounted for in an EM.

New Baseline for Licensing:

Assessment: Cladding rupture from ballooning is not an issue for SiC/SiC cladding. It may be necessary to perform 3-pt./4-pt. bending tests to demonstrate the nature of failure for the cladding to show that cladding failure does not result in dimensional changes leading to flow blockage or large openings leading to fuel dispersal. New cladding rupture limits can then be proposed based on the accumulated data from the tests.

Mechanical Fracturing

Baseline for Current Fuel-System Designs:

Application: Fuel-rod failure during normal operation, AOOs, and DBAs.

SRP Section 4.2: A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. Cladding integrity may be assumed if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature. Other proposed limits may need to be justified. Results from the seismic and LOCA analysis, as presented in Appendix A to SRP Section 4.2, may show that failures by this mechanism may not occur for less severe events.

Technical Regulatory Requirement: Cladding integrity is maintained if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A – GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs.

For DBAs, fuel-rod failure is permitted. However, 10 CFR 50.67 and 10 CFR 100 require the fission-product release from rods exceeding the thermal-margin criteria be limited to meet offsite radioactivity dose limits.

Design Limit: Loss of cladding integrity for applied stresses above 90% of the irradiated yield stress.

Design-Basis Approach: The term "mechanical fracture" refers to a cladding defect caused by an externally applied force, such as a load derived from core-plate motion or a hydraulic load. These loads are bounded by the loads of a SSE and LOCA, and the mechanical fracturing analysis is usually done as a part of the SSE-LOCA loads analysis.

New Baseline for Licensing:

Assessment:

In order to demonstrate post-accident coolability, 3-pt./4-pt. bending tests can be used to demonstrate the failure mechanism of the cladding to show that no major dimensional changes occur leading to coolant channel blockage. However, compared to metals, ceramics show a much wider statistical variation in failure stresses and strains. It may be necessary to account for this by generated a larger failure database to develop a statistical distribution for failure probability calculations.

Fuel Coolability

This section applies to DBAs. For accidents in which severe fuel damage might occur, core coolability must be maintained to meet the requirements of 10 CFR 50 Appendix A GDC 27 and 35 as they relate to control rod insertability and core coolability. Coolability, or a coolable geometry has traditionally implied the fuel assembly retains its rod-bundle geometry with appropriate coolant channels to permit removal of residual heat. The following criteria are presented to verify that coolability is maintained through to extended burnups.

Cladding Embrittlement

Baseline for Current Fuel-System Designs:

Application: Fuel coolability during postulated LOCAs.

SRP Section 4.2: The ECCS performance analysis must satisfy the fuel design criteria specified within 10 CFR 50.46(b). These criteria ensure a coolable core geometry by preserving appropriate post-quench ductility in the fuel-rod cladding. The current criteria require that: (1) the peak cladding temperature remains below 2200°F, and (2) the peak cladding oxidation remains below a 17 percent equivalent cladding reacted (ECR) limit. These criteria were originally developed on the basis of unirradiated Zr-based alloy test specimens. The Zr-based alloy composition, manufacturing process, and in-reactor corrosion alter the post-quench characteristics of the fuel-cladding material. Rulemaking pursuant to 10 CFR 50.46 is planned to implement a performance-based test program that may dictate post-quench performance requirements and provide an acceptable means to establish specific limits for new cladding materials. Future cladding alloys may need to comply with the post-quench performance requirements specified by the new rule and provide the empirical database to support any limits assigned to the new alloy.

Technical Regulatory Requirement: Cladding embrittlement for a LOCA has an acceptance criteria of 1204°C (2200°F) on peak cladding temperature and 17% on maximum cladding oxidation. These criteria represent a SAFDL as defined or referenced in 10 CFR 50 Appendix A – GDC 27 and 35 as they relate to control rod insertability and core coolability for DBAs.

Design Limit: Cladding embrittlement for a LOCA has an acceptance criteria of 1204°C (2200°F) on peak cladding temperature and 17% on maximum cladding oxidation assuming double-sided oxidation in the burst region.

Design-Basis Approach: The design-basis methods utilize fuel performance analysis codes to calculate the maximum clad temperature during the LOCA. The effects of burnup and pellet additives, such as integral burnable absorber or dopant material, are included through either modified material models or penalty factors.

New Baseline for Licensing:

Assessment: SiC/SiC-composite cladding is not affected by oxidation/hydrogen embrittlement. LOCA embrittlement limits on peak cladding temperatures may be relaxed given data to support a change. A new criterion based on fuel temperatures may be more appropriate.

Violent Expulsion of Fuel

Baseline for Current Fuel-System Designs:

Application: Fuel coolability during postulated RIAs, including CREA or CRDA.

SRP Section 4.2: In severe RIAs, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and rod-bundle geometry of the fuel and produce pressure pulses in the primary system.

Fuel-rod thermal-mechanical calculations, employed to demonstrate compliance with Criteria 1 and Criteria 2 below, may need to be based upon design-specific information accounting for manufacturing tolerances and modeling uncertainties using NRC-approved methods, including burnup-enhanced effects on pellet power distribution, fuel thermal conductivity, FGR, and fuel melting temperature.

1. Peak radial average fuel enthalpy must remain below 230 cal/g.
2. Peak fuel temperature must remain below incipient fuel melting conditions.
3. Mechanical energy generated because of: (1) non-molten fuel-to-coolant interaction, and (2) fuel-rod burst, must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
4. No loss of coolable geometry due to: (1) fuel pellet and cladding fragmentation and dispersal, and (2) fuel-rod ballooning.

Technical Regulatory Requirement: GDC-28 defined in 10 CFR Part 50, Appendix A, specifies that reactivity control systems shall be designed to assure that the effects of a postulated reactivity accident result in neither: (1) damage to the RCPB greater than limited local yielding; nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to cause serious impairment of core cooling capability. GDC-28 further specifies that RIAs shall include consideration of rod ejection (unless prevented by positive means), rod drop, steam line rupture, changes in coolant temperature and pressure, and the addition of cold water.

Design Limit: The fuel enthalpy limit is based on the review by NRC staff of the available data (prior to 1974) from the Special Power Excursion Reactor Test (SPERT) and TREAT experimental programs describing the fuel failure consequences following a high rate of reactivity insertion. The review discovered that the potential for prompt rupture of a fuel rod and the rapid heat transfer from finely dispersed molten UO_2 at high fuel energy depositions does exist. Prompt fuel element rupture is defined in RG 1.77 as a rapid increase in internal fuel-rod pressure due to extensive fuel melting, followed by rapid fragmentation and dispersal of cladding into the coolant. The review concluded the failure consequences of UO_2 fuel rods were insignificant for total energy depositions below 300 cal/g for both unirradiated and irradiated fuel rods. As a result, a peak radially averaged fuel enthalpy of 230 cal/g was considered to be a conservative maximum limit to ensure that core damage may be minimal and that both short-term and long-term core cooling capability may not be impaired.

The violent expulsion of fuel design limit specified in SRP Section 4.2 is based on the 230 cal/g peak radially averaged fuel enthalpy limit is applied to both PWR and BWR RIAs.

Design-Basis Approach: The approach to demonstrate compliance to the violent expulsion of fuel design limit for an RIA consists of the following steps:

1. Calculate the maximum energy deposition following a rapid insertion of reactivity using conservative neutron kinetics methods and assumptions,
2. Perform a thermal analysis using the maximum energy deposition to identify the maximum radially averaged fuel enthalpy.

The methods and assumptions used to perform the neutron kinetics calculations are dependent on the specific methodology, fuel design, and fuel management scheme. The areas that require conservative considerations are defined in RG 1.77 for PWRs or RG 1.236 for BWRs.

The results of static core depletion calculations are used to initialize the nuclear parameters and define fuel assembly peaking factors for worst-case conditions. Based on the nuclear parameters listed above, either point, 1D, 1.5D or 3D spatial kinetics methods are used to calculate the core average power pulse transient for the ejection/drop of the highest worth control rod.

The hot spot power transient is calculated from the assembly peaking factors based on the limiting xenon distribution during the entire cycle. This information is input into the thermal heat conduction analysis to calculate the radially average fuel pellet enthalpy. Normally, adiabatic pellet heat-up is assumed in the calculation of the fuel enthalpy to obtain the maximum values.

New Baseline for Licensing:

Assessment: Based on the experimental data from zero and the low-burnup RIA-simulation tests, it is most appropriate to limit the peak pellet temperature to below the fuel melting temperature to mitigate the adverse consequences of fuel-coolant interaction in the unlikely event of dispersal of pellet material. Restricting the fuel enthalpy level to values below that necessary to produce fuel pellet melting will ensure fuel-rod geometry is maintained throughout an RIA event.

Generalized Clad Melting

Baseline for Current Fuel-System Designs:

Application: Fuel coolability during DBAs.

SRP Section 4.2: Generalized (i.e., nonlocal) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in Section B.4.1 are more stringent than the melting criteria are. Therefore, additional specific criteria are not used. However, this may not always be the case for newer alloys or reactor types.

Technical Regulatory Requirement: The requirement to maintain the core in a geometry ensuring control rod insertability and fuel coolability for DBAs is stated in GDC 27 and 35, which are defined in 10 CFR 50 Appendix A. To meet these requirements, fuel coolability criteria should be specified that precludes severe fuel damage mechanisms.

Design Limit: The cladding embrittlement limit defined in Section 3.4.1 is more limiting and is used to ensure that generalized clad melting is avoided during DBAs. The maximum cladding temperature limit for cladding embrittlement is lower than the Zr-based clad melting temperature.

Design-Basis Approach: The maximum cladding temperature is calculated during the analysis of a postulated accident.

New Baseline for Licensing:

Assessment: The methods used to calculate the maximum cladding temperature may need to consider the effect of burnup on the thermal and mechanical properties of the fuel and cladding and the effect of the CRUD layers on the fuel and cladding temperatures.

Fuel-Rod Ballooning

Baseline for Current Fuel-System Designs:

Application: Fuel coolability during DBAs.

SRP Section 4.2: To meet the requirements of 10 CFR 50.46 as it relates to ECCS performance during accidents, the analysis of the core flow distribution may need to account for burst strain and flow blockage caused by ballooning (swelling) of the cladding. RG 1.157 describes acceptable models, correlations, data, and methods that can be used to meet the requirements for a realistic calculation of ECCS performance during a LOCA. Alternatively, Appendix K of 10 CFR Part 50 outlines the acceptable features of a conservative EM to consider burst strain and flow blockage. Burst strain and flow blockage models may need to be based on applicable data to: (1) properly estimate the temperature and differential pressure at which the cladding may rupture (see Section B.3.7), (2) avoid underestimating the resultant degree of cladding swelling, and (3) avoid underestimating the associated reduction in assembly flow area.

The flow blockage model evaluation is provided to the organization responsible for the review of transient and accident analyses for incorporation in the comprehensive ECCS EM to demonstrate the criteria in 10 CFR 50.46(b) are not exceeded. The reviewer also identifies whether the analysis of AOOs and other accidents may need to include fuel-rod ballooning. The possibility of ballooning during an AOO transient or accident increases as the fuel-rod pressure exceeds the system pressure. Those non-LOCA accidents resulting in clad ballooning may need to examine the possibility of DNB propagation resulting from ballooning. The impact of ballooning on non-LOCA accidents may need to not be underestimated. A limit on ballooning (circumferential strain) may be required to prevent DNB propagation for these accidents.

Technical Regulatory Requirement: To meet the requirement of 10 CFR 50.46 as it relates to evaluating ECCS performance during accidents, burst strain and flow blockage caused by ballooning of the cladding may need to be accounted for in the analysis of the core flow distribution

Design Limit: Fuel cladding will balloon under certain combinations of temperature, heating rate, and stress during a LOCA. 10 CFR 50.46 requires that burst strain and flow blockage caused by ballooning of the cladding may need to be accounted for in the analysis of the core flow distribution.

Design-Basis Approach: The design-basis methods utilize cladding deformation and rupture models directly coupled to models for cladding ballooning and flow blockage used in NRC-approved LOCA-ECCS EMs. Other parameters that are important to the LOCA analysis are those input to the analysis from steady-state operation.

New Baseline for Licensing

Assessment: This limit is not appropriate for SiC/SiC-composite as the cladding will not balloon during a LOCA transient. Testing has shown that SiC/SiC-composites do not balloon and burst like Zr alloys. In order to demonstrate post-LOCA coolability, 3-pt./4-pt. bending tests can be used to demonstrate the failure mechanism of the cladding to show that no major dimensional changes occur leading to coolant channel blockage.

Structural Deformation

Baseline for Current Fuel-System Designs:

Application: Fuel coolability during DBAs.

SRP Section 4.2: Acceptance criteria are discussed in SRP Section 4.2, Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" for both a LOCA and an SSE.

1. Loss-of-Coolant Accident:

Two principal criteria apply for the LOCA: (1) the fuel-rod fragmentation must not occur as a direct result of the blowdown loads, and (2) the 10 CFR 50.46 temperature and oxidation limits must not be exceeded. The first criterion is satisfied if the combined loads on the fuel rods and components other than grids remain below the allowable values defined above. The second criterion is satisfied by an ECCS analysis. If combined loads on the grids remain below P(crit), (see Section B.2.1), then no significant distortion of the fuel assembly would occur and the usual ECCS analysis is sufficient. If combined grid loads exceed P(crit), then grid deformation may need to be assumed and the ECCS analysis may need to include the effects of distorted fuel assemblies. An assumption of maximum credible deformation (i.e., fully collapsed grids) may be made unless other assumptions are justified. Control rod insertability is a third criterion that may need to be satisfied. Loads from the worst-case LOCA requiring control rod insertion may need to be combined with SSE loads, and control rod insertability may need to be demonstrated for that combined load. For a PWR, if combined loads on the grids remain below P(crit), as defined above, then significant deformation of the fuel assembly would not occur, and lateral displacement of the guide tubes would not interfere with control rod insertion. If combined loads on the grids exceed P(crit), then additional analysis is needed to show the deformation is not severe enough to prevent control rod insertion.

2. Safe-Shutdown Earthquake:

Two criteria apply to the SSE: (1) fuel-rod fragmentation may need to not occur as a result of the seismic loads, and (2) control rod insertability may need to be assured. The first criterion is satisfied by the criteria in the preceding LOCA discussion. The second criterion may need to be satisfied for SSE loads alone if the preceding LOCA discussion does not require an analysis for combined loads.

Technical Regulatory Requirement: Cladding integrity is maintained if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature represents a SAFDL as defined or referenced in 10 CFR 50 Appendix A - GDC 10, 12, 17, 20, 25, 27, and 35. These criteria require that the reactor core, coolant, control, and protection systems shall be designed to assure appropriate margin to SAFDLs during normal operation or AOOs. For DBAs, fuel-rod failure is permitted. However, 10 CFR 50.67 and 10 CFR 100 require that the fission-product release from rods that exceed the thermal-margin criteria be limited to meet offsite radioactivity dose limits.

Design Limit: Loss of cladding integrity for applied stresses above 90% of the irradiated yield stress.

Design-Basis Approach: Earthquakes and postulated pipe breaks in the RCS would result in external forces on the fuel assembly. The fuel-system coolability should be maintained during such events and the damage should not be so severe as to prevent control rod insertion when required during these low probability accidents.

The design-basis is that the fuel assembly may maintain a geometry that is capable of being cooled under the worst-case accident and that control element insertability may be maintained during an SSE. Analytical methods are used to combine SSE-LOCA loads. These analysis methods include the fuel assembly structural response and also fuel-rod cladding loads. These analyses use site specific input for ground motions.

New Baseline for Licensing:

Assessment: No change in the existing design limit seems to be warranted for implementation of SiC/SiC-composite cladding.

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