July 16, 2020 TP-LIC-LET-0002

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington DC 20555-0001

Subject: TerraPower, LLC - Regulatory Guidance Development Report

On July 1, 2020, TerraPower LLC met with NRC Staff to discuss TerraPower submittals, describe the submittals, and review proposed submittal dates. NRC Staff was provided an overview of the *Advanced Fuel Qualification Methodology Report* which is being developed through a Regulatory Assistance Grant from the DOE. One of the objectives described in the DOE grant was to conduct pre-application interactions with the NRC on the *Advanced Fuel Qualification Methodology Report.* Pre-application interactions will result in a higher-quality report as a result of NRC feedback. Another benefit is the knowledge and experience gained by the NRC on regulatory requirements for metallic fuel as a result of reviewing the report.

For established reactor technologies, the guidance on requirements, format and information required by the NRC to demonstrate compliance is identified in Regulatory Guide (RG) 1.206, *Combined License Applications for Nuclear Power Plants*. In general, information in RG 1.206 is reflected in NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*. Both RG 1.206 and NUREG-0800 were developed primarily for light water reactor (LWR) licensing in the US.

The *Advanced Fuel Qualification Methodology Report* is being developed to provide regulatory guidance for metallic fuel for Sodium Fast Reactors (SFRs.) It will describe methodologies, regulatory criteria and qualification criteria for advanced reactor metallic fuel for SFRs. The final report will include a Fuel Pin Qualification Plan, a Fuel Assembly Qualification Plan and a Regulatory Guidance Development Report. The methodology is expected to help later development of a Fuel Qualification Methodology Topical Report specific to TerraPower fuel. The Fuel Pin Qualification Plan and the Fuel Assembly Qualification Plan are being developed separately and will be completed later.

The Regulatory Guidance Development Report (Enclosure One) describes how to develop and implement steps to identify regulatory requirements, acceptance criteria and compliance approaches for SFR metallic fuel that may need to be addressed in a license application to the NRC. The submittal

includes a generic Regulatory Compliance Plan that documents the implementation of the process for a generic SFR.

This submittal is a White Paper. TerraPower is requesting the NRC to review and evaluate the Regulatory Guidance Development Report and provide preliminary feedback on the approach used to develop regulatory guidance and Regulatory Acceptance Criteria for Advanced Reactor metallic fuel. Factors that may be considered in the review include but are not limited to the following:

- Have the correct set of regulations and principal design criteria been identified?
- Are the regulatory acceptance criteria appropriate?
- Have adequate compliance approaches been identified?

Based on feedback from the July 1, 2020 meeting, TerraPower understands the NRC will perform a preliminary assessment of the Regulatory Guidance Development Report to understand the scope and content. After the preliminary assessment, TerraPower would like to request a follow-up meeting with the NRC to establish the schedule and scope for the complete review. This meeting will be coordinated between TerraPower and NRC Staff.

If you have any questions, please contact me at 423-208-2188 or at pgaillard@terrapower.com. This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

Sincerely,

Peter C. Gaillard

Peter C. Gaillard, PE
Director, Regulatory Affairs
TerraPower, LLC

Enclosure 1: Regulatory Guidance Development Report

Distribution: Mr. Ben Beasley, NRR/DANU/UARL, NRC
Mr. Michael Franovich, NRR/DRA, NRC
Ms. Theresa Lalain, OCHCO/ADHRTF/LTDB
Mr. John Monninger, NRR/DANU, NRC
Mr. John Segala, NRR/DANU/UARP, NRC
Ms. Mallecia Sutton, NRR/DANU/UARL, NRC
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ENCLOSURE ONE

TerraPower®, LLC

White Paper

ADVANCED FUEL QUALIFICATION METHODOLOGY REPORT

Regulatory Guidance Development Report

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1. INTRODUCTION

The Advanced Fuel Qualification Methodology Report is being developed by TerraPower, LLC to provide generic regulatory guidance for qualification of metallic fuel for Sodium-cooled Fast Reactors (SFRs). The report is being developed through a Regulatory Assistance Grant from the Department of Energy, Office of Nuclear Energy. Regulatory Assistance Grants provide direct support for resolving design regulatory issues, regulatory review of licensing topical reports or papers, and other efforts focused on obtaining certification and licensing approvals for advanced reactor designs and capabilities.

The Advanced Fuel Qualification Methodology Report addresses the lack of specific regulatory guidance for metallic fuel for SFRs. The Report will include the Fuel Pin Qualification Plan, the Fuel Assembly Qualification Plan and the Regulatory Guidance Development Report. The Fuel Pin Qualification Plan and the Fuel Assembly Qualification Plan are being developed separately.

The Regulatory Guidance Development Report is the subject of this report and describes how to identify and develop regulatory requirements and compliance approaches applicable for SFR metallic fuel that may need to be addressed in a license application to the Nuclear Regulatory Commission (NRC.) For established reactor technologies, the guidance on requirements, format and information required by the NRC to demonstrate compliance is identified in Regulatory Guide (RG) 1.206, *Combined License Applications for Nuclear Power Plants* [Reference 1]. In general, information in RG 1.206 is reflected in NUREG-0800F, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* [Reference 2]. The Standard Review Plans (SRPs) provide acceptance criteria that applicants can use to demonstrate compliance with regulatory requirements. The SRPs may also provide guidance or expectations for compliance methods used to meet the acceptance criteria. Both RG 1.206 and NUREG-0800 are applicable for light water reactor (LWR) licensing in the US.

2. OBJECTIVES

The primary objective of the Regulatory Guidance Development Report is to create a methodology or process to develop regulatory guidance for advanced reactor metallic fuel. It describes steps to identify regulatory requirements and compliance approaches applicable for SFR metallic fuel that may need to be addressed in a license application to the NRC.

3. METHODOLOGY AND STEPS

The methodology described in this report includes identification of regulatory requirements applicable to metallic fuel qualification, identification of Regulatory Acceptance Criteria (RAC) to ensure compliance with regulatory requirements, and descriptions of compliance approaches based on the RAC and other relevant information from the SRP. The first step is to define the fuel system design under consideration (See Figure 1). The next steps are to identify Acceptance Criteria, develop regulatory requirements and design criteria for SFR fuel, and develop specific compliance approaches applicable for metallic fuel.

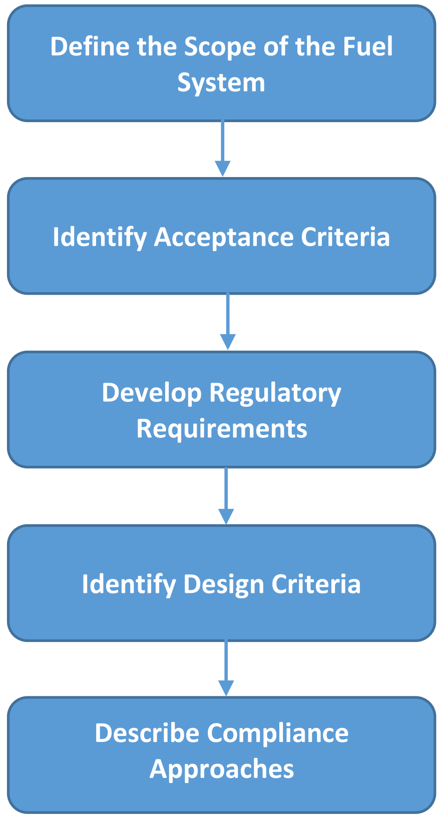

Figure 1 - Fuel Qualification Methodology

4. PROCESS DESCRIPTION

This section describes the steps to define the scope of the fuel system, develop and identify regulatory requirements, develop and identify design criteria, and develop compliance approaches for SFR fuel. Regulatory requirements and design criteria need to be identified so developers, designers and engineers can ensure requirements are appropriately addressed during the design of the fuel system and are documented when developing the Safety Analysis Report (SAR).

The regulatory requirements and guidance on compliance approaches applicable *for LWR fuel* are identified in NUREG-0800 Section 4.2, *Fuel System Design*. RG 1.206, Section C.1.4, *Reactor*, designates the primary information needed to address expectations found in NUREG-0800 Section 4.2.

Regulatory requirements and compliance approaches *for metallic fuel* are based on NUREG-0800 Section 4.2 with modifications to reflect SFR characteristics. NUREG-0800 Section 4.3, *Nuclear Design*, addresses the nuclear design of the fuel assemblies, control systems, and reactor core. Section 4.4, *Thermal and Hydraulic Design*, addresses the thermal and hydraulic design of the core and the reactor coolant system. The requirements in Section 4.3 and Section 4.4 are beyond the scope of this report but the methodology described in this report may be applied to those sections to develop regulatory requirements and compliance approaches applicable to different fuel types.

Define the Scope of the Fuel System

The first step in the process is to define the scope of the fuel system being evaluated. The Scope section should be one or two paragraphs that describes the scope and defines the SFR fuel system design for which regulatory requirements and compliance approaches will be developed. This provides the basis to ensure:

- The fuel system is not damaged during any condition of normal operation, including the effects of anticipated operational occurrences.
- The number of fuel pin failures predicted for postulated accidents is not underestimated.
- Fuel coolability will be maintained during postulated accidents.
- Fuel system damage during postulated accidents will not prevent reactivity control/standby rod insertion when required.

NUREG-0800 describes the fuel system for an LWR as “arrays (assemblies or bundles) of fuel rods, including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods.” NUREG-0800 Section 4.2 also discusses the reactivity control elements of the control rods that extend from the coupling interface of the control rod drive mechanism into the core.

For a generic SFR, a definition of a fuel system has been developed to identify the fuel system design that needs to be addressed. The Generic Regulatory Compliance Plan provides the following description for the generic SFR fuel system:

> *The fuel system consists of the fuel assemblies, reflector assemblies, shield assemblies, and reactivity control assemblies. This section discusses the elements of the reactivity control assemblies in the core region below the coupling interface with the control rod drive mechanism. [NOTE: Each advanced reactor developer would provide additional details as appropriate for their plant-specific design features.]*

Regulatory Acceptance Criteria (RAC)

Fuel design RAC specify design, analysis, programmatic, testing, documentation, or other requirements necessary to ensure the fuel design complies with applicable regulatory requirements and expectations. The RAC ensure compliance with regulatory requirements and design criteria applicable to metallic fuel and establish functional and performance design requirements. The RAC also provide guidance for acceptable compliance approaches used to demonstrate that the RAC are met. RAC identified by advanced reactor developers should be provided to the NRC for concurrence prior to submittal of the SAR.

Each RAC contains the following information:

- Identification Number (ID)
- Acceptance Criteria
- Parent Requirement

- Basis
- Compliance description
- Compliance Specific Considerations
- Additional Information

RAC Identification Numbers (e.g., RAC 4.2-1) used in this report correspond to NUREG-0800 section numbers. We are aware of industry initiatives including the Licensing Modernization Project and the Technology Inclusive Content of Application Project that when implemented may change the format and content of future licensing submittals. The decision to identify RAC using this scheme was selected for convenience.

Acceptance Criteria

The fuel design Acceptance Criteria are analogous to NUREG-0800 Acceptance Criteria in SRP Section 4.2. All Acceptance Criteria in SRP Section 4.2 were identified and evaluated. The Acceptance Criteria may be used-as-is, modified or removed as appropriate. New Acceptance Criteria may be added if appropriate for a fuel design. In some cases, Acceptance Criteria for metallic fuel will differ from the NUREG-0800 Acceptance Criteria due to inherent differences between Advanced Reactor fuel and LWR technology. Every applicable Acceptance Criteria identified as a result of the NUREG-0800 Section 4.2 review is listed in the Regulatory Compliance Plan.

Every Acceptance Criteria has a Parent Requirement and a Basis. Meeting the Acceptance Criteria ensures the fuel design complies with applicable regulatory requirements and design criteria called Parent Requirements. The Basis identifies the applicable part of NUREG-0800 from which the Acceptance Criteria was derived.

Applicable Regulatory Requirements and Design Criteria

Specific SFR regulatory requirements, i.e., 10 CFR requirements, are based on an assessment of current LWR regulatory requirements identified and described in NUREG-0800, Section 4.2, *Fuel System Design*. The regulatory requirements are revised and modified to reflect SFR metallic fuel characteristics. The description developed and provided for each specific SFR regulatory requirement generally does not need to repeat the entire 10 CFR requirement. The SFR regulatory requirement description should focus on the aspect of the regulatory requirement addressed in the RCP (similar to the approach used in Section II of the NUREG-0800 SRPs).

General Design Criteria (GDC) are defined in 10 CFR 50 Appendix A, *General Design Criteria for Nuclear Power Plants*. 10 CFR 50 Appendix A acknowledges that GDC may not be necessary or appropriate for some advanced reactor designs. In some cases, additional or different GDC may be required for various advanced reactor designs.

The GDC establishes minimum requirements for Principal Design Criteria (PDC) for LWRs. RG 1.232, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors* [Reference 3] describes NRC guidance for adapting 10 CFR 50 Appendix A GDC for non-light-water reactor (non-LWR) designs. RG 1.232 also describes the NRC's proposed guidance for modifying and supplementing the GDC to develop SFR design criteria (called Sodium Fast Reactor-Design Criteria or SFR-DC). Section 2, *Applicable Regulatory Requirements* of Attachment One, identifies regulatory requirements

and SFR-DC applicable for metallic fuel design and they are summarized in Table 1, *Regulatory Requirements and Design Criteria*.

Non-LWR reactor designers, applicants, and licensees may develop specific PDC for their designs based on SFR-DC. The specific PDC provide a safety-basis equivalent to the requirements of 10 CFR 50 Appendix A GDC. They are high-level safety criteria that establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. The general approach for developing PDCs is to only make changes necessary to Appendix A GDC or SFR-DC for application to a specific SFR design.

Each reactor developer should review RG 1.232 SFR-DC to develop PDC based on their specific SFR characteristics and unique design features for a particular reactor design. Requirements that are not applicable may be modified or removed. A substantial number of the PDC requirements may use the same wording as found in 10 CFR 50 Appendix A or RG 1.232. However, a number of requirements may be different or new based on the fundamentals of the SFR design.

The regulatory requirements and SFR-DC applicable to metallic fuel are identified in Attachment One, Section 2 and summarized in Table 1. SFR-DC are cited because PDC cannot de designated for this generic assessment.

Table 1 – Regulatory Requirements and Design Criteria

SFR-DC 10	<u>Reactor design</u> - as it relates to assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.
SFR-DC 26	<u>Combined reactivity control systems capability</u> - as it relates to the design of the reactivity control system with appropriate margin such that the reactivity control system is capable of controlling reactivity during postulated accident conditions.
SFR-DC 35	<u>Emergency core cooling</u> - as it relates to assuring continuous effective cooling is provided following postulated accidents such that fuel damage that could interfere with continued effective core cooling is prevented and the design conditions of the primary system boundary are not exceeded.
SFR-DC 2	<u>Design bases for protection against natural phenomena</u> - as it relates to designing structures, systems, and components important to safety to withstand an appropriate combination of loads from natural phenomena and accident conditions.
10 CFR 100	As it relates to determining the acceptability of a reactor site based on calculating the exposure to an individual as a result of fission product releases to the environment following a major accident.
10 CFR 50.62	As it relates to the design of the standby shutdown system as an alternate means to shut down the core to provide a reduction in risk from anticipated transients without scram (ATWS) events.

Compliance Approaches

Demonstrating compliance is an important task to support licensing. Compliance approaches including Compliance Descriptions, Compliance Specific Considerations and Additional Information are described in each RAC. They may specify design, analysis, programmatic, testing, documentation, or other requirements necessary to ensure the plant design complies with applicable regulatory requirements and expectations.

The Compliance Description can be a simple one-sentence description, i.e., “establish criteria based on test data, analysis, engineering evaluations, or industry standards.” Compliance Specific Considerations may describe recommendations, acceptable approaches, important aspects, key factors, major assumptions, or other guidance provided on how compliance with the RAC may be accomplished. Input for the Compliance Description and the Compliance Specific Considerations may be based on information in the SRP modified as appropriate for the SFR design.

Additional Information may be added to provide other relevant guidance or references used as a basis for RAC or compliance methods. This may include information not necessarily addressed in the SRP including information from previous SFR applications or staff experience that may be beneficial in understanding the acceptance criteria or developing compliance methods.

Generic Regulatory Compliance Plan

The results from implementing the process described in this report may be documented in a Regulatory Compliance Plan (RCP). Information in an RCP may include:

- Scope of the system to be evaluated including as appropriate the systems, components, analyses, data, or other information to be evaluated in the RCP
- Specification of applicable regulatory requirements or design criteria
- Identification of design requirements (directly or by reference to codes and standards)
- Identification of design and safety analyses
- Identification of test programs that may be required
- Identification of design, construction, or operation procedural control programs

Attachment One, *Generic Regulatory Compliance Plan, Fuel System Design*, documents and describes generic regulatory requirements for SFR metallic fuel identified as a result of implementing the process described in this report. The regulatory requirements are similar to requirements described in NUREG-0800, Section 4.2. Where applicable, notes have been added where each advanced reactor developer might provide additional details appropriate for their plant-specific design features.

5. ACRONYMS

ATWS	Anticipated Transients Without Scram
GDC	General Design Criteria

LWR	Light Water Reactor
NRC	Nuclear Regulatory Commission
PDC	Principal Design Criteria (PDC)
RAC	Regulatory Acceptance Criteria
RCP	Regulatory Compliance Plans
RG	Regulatory Guide
SAR	Safety Analysis Report
SFR	Sodium Fast Reactor
SFR-DC	Sodium Fast Reactor-Design Criteria
SRP	Standard Review Plans

6. REFERENCES

1. *Combined License Applications for Nuclear Power Plants,* Regulatory Guide (RG) 1.206, June 2007.
2. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800.
3. *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*, RG 1.232, April 2018.

ATTACHMENT ONE

Generic Regulatory Compliance Plan

Fuel System Design

ATTACHMENT ONE

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1. SCOPE

This Regulatory Compliance Plan (RCP) identifies the regulatory requirements addressed in Section 4.2, *Fuel System Design* of Safety Analysis Reports (SARs) prepared for [*PlantName*]. This RCP also provides the Regulatory Acceptance Criteria (RAC) established to ensure compliance with the requirements addressed in the SAR section. The scope and content of each section of the [*PlantName*] SAR are described in Reference 1. [*NOTE: Update Reference 1 or remove citation*].

The fuel system consists of the fuel assemblies, reflector assemblies, shield assemblies, and reactivity control assemblies. [*NOTE: Additional details should be added as appropriate for plant specific design features.*] This section discusses the elements of the reactivity control assemblies in the core region below the coupling interface with the control rod drive mechanism.

This section describes requirements for fuel system design necessary to ensure:

1. The fuel system is not damaged during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

2. The number of fuel pin failures predicted for postulated accidents is not underestimated.

3. Fuel coolability will be maintained during postulated accidents.

4. Fuel system damage during postulated accidents will not prevent reactivity control rod insertion when required.

2. APPLICABLE REGULATORY REQUIREMENTS

Regulatory Requirements addressed in the SAR are identified and provided in Reference 2. Principal Design Criteria (PDC) are based on generic sodium fast reactor design criteria (SFR-DC) provided in Reference 3. Regulatory requirements that are applicable within the scope of and specifically addressed in the SAR section are as follows:

[*NOTE: If plant specific PDC are developed, replace citations to the generic SFR design criteria (SFR-DC) with the appropriate PDC in the remainder of this RCP. Add a reference for the plant specific PDC in Section 4 and cite the reference in the above paragraph.*]

SFR-DC 10	Reactor design - as it relates to assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.
SFR-DC 26	Combined reactivity control systems capability - as it relates to the design of the reactivity control system with appropriate margin such that the reactivity control system is capable of controlling reactivity during postulated accident conditions.
SFR-DC 35	Emergency core cooling - as it relates to assuring continuous effective cooling is provided following postulated accidents such that fuel damage that could interfere with continued effective core cooling is prevented and the design conditions of the primary system boundary are not exceeded.
SFR-DC 2	Design bases for protection against natural phenomena - as it relates to designing structures, systems, and components important to safety to withstand an appropriate combination of loads from natural phenomena and accident conditions.
10 CFR 100	As it relates to determining the acceptability of a reactor site based on calculating the exposure to an individual as a result of fission product releases to the environment following a major accident.

10 CFR 50.62	As it relates to the design of the standby shutdown system as an alternate means to shut down the core to provide a reduction in risk from anticipated transients without scram (ATWS) events.

3. REGULATORY ACCEPTANCE CRITERIA

The following Regulatory Acceptance Criteria (RAC) are established to ensure compliance with the regulatory requirements identified in Section 2.

ID	**RAC 4.2-1 Fuel System Damage Criteria**
Acceptance Criterion	Fuel system damage criteria shall be established for normal operation, including AOOs, to ensure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.
Parent Requirement	SFR-DC 10
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A
Compliance Description	Establish fuel system design criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Fuel system damage criteria should be included for all known damage mechanisms. During operation, the fuel system assemblies may be subject to mechanical stresses due to processes such as fuel handling and loading, power and thermal gradients, irradiation, flow-induced vibration and fretting, and creep deformation. The fuel system design should account for the impact of these factors on the integrity of the fuel system assemblies. Fuel system damage criteria should ensure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel system damage criteria should consider high burnup effects based on irradiated material properties data. The effects of fast neutrons on the metallurgical properties and structural stability of the fuel system assemblies should be considered. For metallic fuel, the redistribution of fuel alloying elements and fission products should also be considered. Design-basis limits and associated specified acceptable fuel design limits (SAFDLs) should be assessed to determine whether they remain applicable for new fuel designs (including the introduction of new materials) or for changes in the planned operating conditions (temperature, burnup, and power).
Additional Information	Specified acceptable fuel design limits (SAFDLs) are established to provide assurance that the fuel system is not damaged and fuel pin failures do not occur during normal operation or as a result of anticipated operation occurrences. Fuel system damage means that fuel system dimensions are outside operational tolerances or the functional capabilities of the fuel system are reduced below those assumed in the safety analysis.

ID	**RAC 4.2-1.1**
Acceptance Criterion	Stress, strain, or loading limits for all fuel system components shall be established.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.i
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Fuel system components consist of fuel pins, reactivity control absorber pins, fuel and reactivity control assembly ducts, and other fuel system structural members such as reflector and shield assemblies. Stress limits that are obtained by methods similar to those given in Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME) are acceptable. Other proposed limits must be justified.
Additional Information	Regarding Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME), other criteria may apply for the sodium fast reactor (SFR), while some of the light water reactor (LWR) criteria do not apply. The cladding irradiation effects of concern for SFRs are creep and swelling, while for LWRs it is growth.

ID	**RAC 4.2-1.2**
Acceptance Criterion	The cumulative number of strain fatigue cycles on all fuel system components shall be significantly less than the design fatigue lifetime.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.ii
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Fuel system components consist of fuel pins, reactivity control absorber pins, fuel and reactivity control assembly ducts, and other fuel system structural members such as reflector and shield assemblies. An acceptable limit should be based on appropriate data and include a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. Other proposed limits must be justified.
Additional Information	Contributions of high temperature creep to fatigue usage should be addressed (refer to ASME Section III, Div. 5).

ID	**RAC 4.2-1.3**
Acceptance Criterion	Limits on fretting wear at contact points on all fuel system components shall be established.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.iii
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Fuel system components consist of fuel pins, reactivity control absorber pins, fuel and reactivity control assembly ducts, and other fuel system structural members such as reflector and shield assemblies. Fretting wear tests and analyses that demonstrate compliance with this design basis should account for pin interaction with the wire wrap, as well as pin-to-pin and pin-to-duct interactions. Stress, strain, and fatigue limits should presume the existence of the allowable fretting wear.
Additional Information	None.

ID	**RAC 4.2-1.4**
Acceptance Criterion	Limits on erosion and the buildup of corrosion products shall be established for all fuel system components.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.iv
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Fuel system components consist of fuel pins, reactivity control absorber pins, fuel and reactivity control assembly ducts, and other fuel system structural members such as reflector and shield assemblies. Limits on erosion and corrosion product buildup should be established based on mechanical testing to demonstrate that each component maintains acceptable strength and ductility. Stress, strain, and fatigue limits should presume the existence of the allowable erosion and corrosion product buildup.
Additional Information	For an SFR, oxidation and hydriding are not a concern. However, corrosion and erosion are concerns. Crud is a phenomena of concern for LWR fuel, but not for SFR fuel.

ID	**RAC 4.2-1.5**
Acceptance Criterion	Limits on cladding damage (wastage) due to fuel-cladding chemical interaction (FCCI) shall be established.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.iv FCCI is an SFR phenomenon resulting in cladding wastage (thinning), similar to the LWR phenomena oxidation and hydriding discussed in Criterion 1.A.iv.
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	The mechanisms of FCCI for metallic fuel are the fuel constituent migration, cladding attack by rare-earth fission products, and interdiffusion between the fuel alloy and cladding, all eventually resulting in reduction of effective cladding wall and formation of a low-melting point eutectic. Since this process is temperature and burnup dependent, the criterion for limiting FCCI can be established by limiting the maximum temperature of the fuel-cladding interface to a specified value as a function of burnup, temperature, and the time spent at high temperature. Alternatively, the criterion can be in terms of the maximum allowed cladding thinning (degraded reaction zone and eutectic penetration) due to FCCI. Stress, strain, and fatigue limits should presume the existence of the allowable cladding thinning due to FCCI.
Additional Information	None.

ID	**RAC 4.2-1.6**
Acceptance Criterion	Limits on dimensional changes, such as pin bowing, assembly duct bowing, pin swelling, and assembly duct dilation, shall be established to ensure that fuel, reflector, and shield assembly dimensions remain within operational tolerances or to prevent a situation where thermal hydraulic or neutronic design limits are exceeded.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.v
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Deformation of fuel pins or fuel assembly ducts should be considered in ensuring that other fuel design limits are met and adequate fuel cooling is maintained. Limits on fuel system dimensional changes may be necessary to meet thermal hydraulic and neutronic design criteria.
Additional Information	The cladding irradiation effects of concern for SFRs are creep and swelling, while for LWRs it is growth. Fuel bundle-to-duct interaction due to irradiation creep and swelling may affect thermal/hydraulic performance as well as structural integrity of fuel bundle and duct.

ID	**RAC 4.2-1.7**
Acceptance Criterion	Limits on dimensional changes, such as absorber pin bowing, assembly duct bowing, absorber pin swelling, and assembly duct dilation, shall be established to ensure that reactivity control assembly dimensions remain within operational tolerances and to prevent interference that may impact control rod insertability.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.v
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Deformation of the absorber pins or assembly ducts should not affect the capability for the insertion of control rods for the safe shutdown of the reactor. If interference is determined to be possible, tests are needed to demonstrate that control rod insertability performance is consistent with assumptions used in safety analyses. Additional in-reactor surveillance (e.g., insertion times) may also be necessary for new designs (changes in dimensions or materials) to demonstrate satisfactory performance.
Additional Information	Shadow corrosion is an LWR concern, but is not an SFR concern. Excessive gaps between ducts may cause the reactivity swing or flow induced vibration of the inner assembly.

ID	**RAC 4.2-1.8**
Acceptance Criterion	Design limits on fuel pin and reactivity control absorber pin internal pressure for normal operation and AOOs shall be established, or alternatively, pin internal pressure shall be explicitly assessed in analyses demonstrating compliance with fuel system damage criteria, failure criteria, or coolability criteria that may be affected by pin internal pressure.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criteria 1.A.vi
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Potential fuel pin failures or impact on coolability due to fuel pin overpressure must be addressed in analyses of AOOs and postulated accidents.
Additional Information	For normal operation, cladding liftoff may be a concern for oxide fuel, but is not an issue for metallic fuel with sodium bonding. Reorientation of hydrides is not a phenomena of concern for SFR fuel types.

ID	**RAC 4.2-1.9**
Acceptance Criterion	The worst-case hydraulic loads for normal operation and AOOs shall not exceed the holddown capability of a fuel, reflector, or shield assembly.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.vii
Compliance Description	Perform analyses to demonstrate holddown capability.
Compliance Specific Considerations	Unseating a fuel assembly may challenge the ability to cool the fuel assembly. Wear and local deformation on the inlet nozzle, receptacle, and piston rings should be within operational tolerances
Additional Information	None.

ID	**RAC 4.2-1.10**
Acceptance Criterion	The worst-case hydraulic loads for normal operation and AOOs shall not exceed the holddown capability of a reactivity control assembly.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.vii
Compliance Description	Perform analyses to demonstrate holddown capability.
Compliance Specific Considerations	Unseating a reactivity control assembly may challenge the ability to insert control rods and the ability to cool the control rods.
Additional Information	None.

ID	**RAC 4.2-1.11**
Acceptance Criterion	Design limits for the mechanical and neutronic lifetimes for reactivity control assemblies shall be established to ensure that control rod reactivity and insertability are maintained.
Parent Requirement	RAC 4.2-1
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.viii
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Mechanical and neutronic lifetimes need to be calculated using acceptable methods. The potential impact of the loss of absorber material should be addressed if leachable materials are used. Safety analyses must specifically account for the reduction in neutron-absorbing capabilities in the control rods during reactor operation. Inner and outer duct bow behaviors should be analyzed. Wear characteristics of components should be evaluated.
Additional Information	Wear pads should be used at both ends of the inner assembly to prevent excessive friction forces between ducts..

ID	**RAC 4.2-2 Fuel Pin Failure Criteria**
Acceptance Criterion	Fuel pin failure criteria shall be established for all failure mechanisms that may result in the loss of fuel integrity (cladding breach) during normal operation, AOOs, and postulated accidents.
Parent Requirement	SFR-DC 10 10 CFR 100
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.B
Compliance Description	Establish fuel pin failure criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	To meet the requirements of SFR-DC 10 as it relates to SAFDLs for normal operation and AOOs and of 10 CFR 100 as it relates to fission product release during postulated accidents, fuel pin failure criteria should be provided for all known failure mechanisms. Although it is impossible to avoid all fuel pin failures and cleanup systems are installed to handle a small number of leaking pins, the design must ensure that fuel does not fail as a result of specific causes during normal operation and AOOs. Fuel pin failures are permitted during postulated accidents, but they must be accounted for in the radiological dose analysis. Fuel pin failures can be caused by overheating of cladding, overheating of fuel slug, deformation of cladding due to mechanical loads, mechanical fracturing of cladding due to external loads, and cladding wastage. When applicable, the fuel pin failure criteria should consider high burnup effects based on irradiated material properties data. The effects of fast neutrons on the metallurgical properties and structural stability of the fuel, blanket, and control assemblies should be considered. For metallic fuel, the redistribution of fuel alloying elements and fission products should also be considered.
Additional Information	Specified acceptable fuel design limits (SAFDLs) are established to provide assurance that the fuel system is not damaged and fuel pin failures do not occur during normal operation or as a result of anticipated operation occurrences. Fuel pin failure is defined as a breach of the first fission product barrier (the cladding), resulting in a potential uncontrolled release of radioactivity material to the coolant. Hydriding is an LWR fuel phenomena and does not occur in fuel types for an SFR. Cladding collapse is an LWR fuel phenomena and does not occur in SFRs.

ID	**RAC 4.2-2.1**
Acceptance Criterion	Fuel system design limits shall be established and used for the prediction of fuel pin failure due to overheating of the cladding.
Parent Requirement	RAC 4.2-2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.B.iii
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	For typical fuel pin linear power generation rates for SFRs, preventing sodium boiling is sufficient to ensure that cladding overheating due to a deficient cooling mechanism can be avoided during normal operation and AOOs. For metallic fuel, the maximum cladding temperature during normal operation and AOOs should be less than the minimum temperature for eutectic liquefaction at the fuel-cladding interface. At temperatures above the eutectic liquefaction threshold, the time scales for eutectic penetration into the cladding are comparable to the time scales of some postulated accidents. The minimum temperature for eutectic liquefaction can be exceeded for short times without excessively damaging fuel pins. Design limits should be established on cladding temperature during postulated accidents to prevent or minimize clad damage due to eutectic liquefaction penetration. Potential cladding damage (loss of clad thickness) from any eutectic liquefaction must be considered in the assessment of potential cladding failure due to other phenomena (such as cladding stress and strain).
Additional Information	Traditional practice for LWRs assumes that failures will not occur due to overheating of the cladding if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. The thermal margin criteria are measures of margin to the degradation in heat removal capability due to departure from nuclear boiling for PWRs or dryout (steam blanketing) for BWRs. These phenomena are not expected to occur in an SFR during normal operation, AOOs, or postulated accidents; therefore, DNBR or CPR are not applicable for SFRs. For SFR fuel, cladding may fail prior to sodium boiling. Therefore, preventing sodium boiling may not be the most limiting concern for cladding overheating.

ID	**RAC 4.2-2.2**
Acceptance Criterion	Fuel system design limits shall be established and used for the prediction of fuel pin failure due to overheating of the fuel slug.
Parent Requirement	RAC 4.2-2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.B.iv
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Traditional regulatory practice has been to assume that cladding failure will occur if fuel melting takes place. For normal operation and AOOs, fuel melting is not permitted. For postulated accidents, the total number of pins that experience fuel melting should be assumed to fail for radiological dose calculation purposes. The assumption that fuel melting results in cladding failure is conservative. The analysis should be performed for the maximum linear heat generation rate 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 (such as constituent redistribution, swelling, accumulation of fission products, release of fission gas to the fuel pin plenum, and other changes in the microstructure of the fuel). The fuel melting criterion was established to ensure that axial or radial relocation of molten fuel would neither allow molten fuel to contact the cladding nor produce local hot spots. Avoiding localized interior fuel melting prevents stress on the cladding from fuel radial expansion due to the large volume increase associated with melting (not a significant issue for metallic fuel due to reduced resistance to axial expansion). Avoiding fuel melting can also preclude cladding eutectic penetration for metallic fuel since the rate of eutectic formation is significantly higher at temperatures that lead to fuel melting. In addition to protecting the cladding from failure, avoiding fuel melting prevents fuel relocation that might lead to unacceptable reactivity changes.
Additional Information	None.

ID	**RAC 4.2-2.3**
Acceptance Criterion	Fuel system design limits shall be established and used for the prediction of fuel pin failure (loss of cladding integrity) due to deformation of the cladding from mechanical loads.
Parent Requirement	RAC 4.2-2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criteria 1.B.v, vi, vii
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Major mechanical loadings on the cladding that may lead to deformation and cladding failure are: • Fuel cladding mechanical interaction (FCMI) due to differential fuel and cladding expansion • Fuel pin internal pressure from increased temperature and fission product gas release from the fuel • Thermal loadings due to temperature gradients and changes during events • Interaction forces between fuel pins and fuel assembly duct due to different thermal expansion and swelling The fuel system design should address potential failures from deformation due to the above loads. Design limits on maximum allowed cladding stress and strain should be established as necessary. The design evaluations assessing the margin-to-failure for cladding deformation should account for all effects that may occur during irradiation, including any fuel-cladding mechanical and chemical interactions, increases in fuel pin internal pressure, and changes in cladding mechanical properties (strength, creep, and stress relaxation). For oxide fuel, stress-driven failures by FCMI are a major concern and are addressed by a limit on uniform strain of the cladding to a specified value. In this context, uniform strain (elastic and inelastic) is defined as the steady-state (creep/swelling) and transient-induced deformation with gauge lengths corresponding to cladding dimensions. Mechanical testing must demonstrate that the irradiated cladding ductility is well within the specified strain limit. For metallic fuel, the fuel is designed with a low smear density to accommodate the initial fuel swelling and achieve a high burn-up. The typically limiting performance issue for metallic fuel is creep rupture of cladding with the cladding temperature, fission-gas-plenum pressure, and cladding irradiated performance properties being the key factors. Creep rupture can be accelerated by cladding wastage due to FCCI. Several different design criteria for cladding failure due to thermal creep have been used for SFRs. Thermal creep criteria should address cladding temperature, pressure, time duration, and embrittlement. Fast reactor fuels are typically designed to reach higher burnup than LWR fuels to take advantage of higher initial fissile loading and the “breed and burn” characteristics unique to the fast neutron spectrum. Furthermore, fuel swelling is expected to be greater in fast spectrum. However, for both the oxide and metallic fuel, gaseous swelling is not an issue at high burnup since fission gas release is very

ID	**RAC 4.2-2.3**
	high after the initial few percent burnup. Solid fission product swelling is the cause of the clad straining at high burnup. Thermal expansion has the same magnitude at low or high burnup cases. Therefore, FCMI analyses of cladding strain for AOOs and postulated accidents should apply approved fuel thermal expansion, solid fission product induced fuel swelling, and fuel creep (especially for metallic fuel) models, as well as irradiated cladding properties. The sudden increase in fuel enthalpy from a reactivity initiated accident (RIA) can result in oxide fuel failure due to FCMI. FCMI is not a major concern for metallic fuel with a softer matrix that is prone to creep. Interaction between fuel pins and the fuel assembly duct due to different thermal expansion and swelling could result in additional stress on the cladding and potentially cause failure of the cladding. The fuel assembly design should ensure that pin-duct interaction does not result in fuel failure during normal operation or AOOs. Design limits on fuel assembly exposure, fuel pin total strain, or other phenomena should be established as necessary to prevent fuel pin failure due to pin-duct interaction during normal operation and AOOs. Design limits to preclude fuel failure or criteria to conservatively estimate fuel failure during postulated accidents shall be established as necessary. Cladding bursting or ballooning from pin internal pressure during postulated accidents should be evaluated and addressed in terms of impact on predicted fuel assembly cooling and radiological consequences. The effect of the steady-state and transient cladding wastage is addressed by assuming that the affected cladding thickness provides no strength for applied loads.
Additional Information	None.

ID	**RAC 4.2-2.4**
Acceptance Criterion	Fuel system design limits shall be established and used for the prediction of fuel pin failure (loss of cladding integrity) due to mechanical fracturing from externally applied forces.
Parent Requirement	RAC 4.2-2 SFR-DC 2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.B.viii
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	A mechanical fracture refers to a failure in a fuel pin caused by an externally applied force such as a hydraulic load or a load derived from fuel support structure motion. Earthquakes and postulated accidents result in external forces on the fuel assembly. To meet the requirements of SFR-DC 2 as it relates to combining loads, an appropriate combination of loads from natural phenomena and accident conditions must be made. Cladding integrity may be assumed if the applied stress is less than 90 percent of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. See Appendix A for discussion of methods for assessing structural deformation due to external forces.
Additional Information	None.

ID	**RAC 4.2-2.5**
Acceptance Criterion	Fuel system design limits established and used for the prediction of fuel pin failure (loss of cladding integrity) shall address the effects of cladding wastage.
Parent Requirement	RAC 4.2-2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.B
Compliance Description	Establish criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Cladding wastage is defined as the portion of the original cladding wall that is damaged and no longer assumed to carry applied loads. The mechanisms that contribute to cladding wastage include: • Solid state diffusion of fuel constituents and fission products in to the cladding due to FCCI • Eutectic liquefaction at the fuel-cladding interface • Sodium-cladding corrosion • Sodium erosion of cladding • Fretting of cladding due to wear at contact points with other structures With the exception of eutectic liquefaction, the mechanisms that contribute to cladding wastage are slow and occur primarily during normal operation; the incremental wastage during AOOs and postulated accidents is negligible and design limits established for normal operation remain applicable for AOOs and postulated accidents. Eutectic penetration of the cladding during postulated accidents is limited by design limits on the maximum-allowed cladding temperature during postulated accidents. The damage (loss of clad thickness) from allowed cladding wastage during normal operation and potentially increased damage due to eutectic penetration during postulated accidents must be addressed in the assessment of potential cladding failure due to other phenomena (such as cladding stress and strain). The effect of the steady-state and transient cladding wastage is addressed by assuming that the affected cladding thickness provides no strength for applied loads. Fretting wear between wrap wire and cladding may cause localized wastage or penetration of fuel claddings. The fretting wear characteristics should be identified by testing or numerical model considering wrap wire configurations such as wire diameter, tension, or pitch.
Additional Information	None.

ID	**RAC 4.2-3 Fuel Coolability Criteria**
Acceptance Criterion	Fuel assembly criteria shall be established for all severe damage mechanisms that may occur during postulated accidents to ensure that the fuel assembly geometry retains adequate coolant flow channels to permit removal of residual heat.
Parent Requirement	SFR-DC 35
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C
Compliance Description	Establish fuel coolability criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Coolability, or a coolable geometry, means that the fuel assembly retains its geometry with adequate coolant channels to permit removal of residual heat. During postulated accidents, fuel failure (loss of cladding integrity) may occur as long as long as a coolable geometry is maintained for the fuel assembly. Fuel coolability criteria should be provided for all known mechanisms that can result in a reduction of coolability. The effects of fast neutrons on the metallurgical properties and structural stability of the fuel, blanket, and control assemblies should be considered. For metallic fuel, the redistribution of fuel alloying elements and fission products should also be considered. Reduction of fuel coolability can result from significant cladding damage, generalized cladding melting, fuel pin (cladding) ballooning, fuel melting and relocation, fuel assembly structural deformation, and fuel assembly liftoff. Excessive debris accumulation in the fuel assembly can result in a reduction of coolability.
Additional Information	Fuel coolability means the fuel assembly retains its pin-bundle geometry with adequate coolant channels to permit removal of residual heat. For SFRs, control rod insertability criteria and core coolability criteria are not as strongly coupled as for LWRs due to the use of separate fuel assemblies and control assemblies for SFRs.

ID	**RAC 4.2-3.1**
Acceptance Criterion	Fuel system design limits shall be established to ensure that cladding stress and strain during postulated accidents do not result in significant cladding damage that might prevent adequate core cooling.
Parent Requirement	RAC 4.2-3
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C.i
Compliance Specific Considerations	During postulated accidents, the loading on the cladding may increase significantly due to temperature gradients, FCMI, and pin internal gas pressure. The transient performance characteristics of the cladding material depend on specific stainless-steel cladding alloy composition, manufacturing process, and in-reactor irradiation. Cladding wastage should be addressed in establishing the design limits or in the evaluations to demonstrate compliance with the design limits. For metallic fuel, loss of cladding due to eutectic liquefaction during postulated accidents should be addressed or, alternatively, design limits on cladding temperature during postulated accidents should be established to limit eutectic liquefaction to ensure a coolable geometry is preserved.
Additional Information	None.

ID	**RAC 4.2-3.2**
Acceptance Criterion	The maximum temperature of the cladding during postulated accidents shall be less than the melting temperature of the cladding.
Parent Requirement	RAC 4.2-3
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C.iii
Compliance Description	Establish fuel coolability criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	Generalized (i.e., nonlocal) melting of the cladding could result in the loss of fuel assembly pin-bundle geometry. For SFRs, other criteria will generally be more limiting and prevent the cladding from reaching the melting temperature during postulated accidents. However, this may not always be the case for different cladding alloys or reactor designs. The effects of high burnup and fast neutrons on the mechanical and thermal properties of the cladding should be considered.
Additional Information	None.

ID	**RAC 4.2-3.3**
Acceptance Criterion	Evaluations of fuel assembly temperatures to demonstrate core coolability must account for the effects on core flow distribution and the potential for flow blockage caused by ballooning (swelling) of the cladding during postulated accidents.
Parent Requirement	RAC 4.2-3
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C.iv
Compliance Description	Establish fuel coolability criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	The analysis of the core flow distribution must account for burst strain and flow blockage caused by thermal-creep-assisted ballooning (swelling) of the cladding. Acceptable models, correlations, data, and methods that can be used to meet the requirements for a realistic calculation of reactor response during a postulated accident should be provided. Burst strain and flow blockage models must be based on applicable data to (1) properly estimate the temperature and differential pressure at which the cladding will rupture, (2) avoid underestimating the resultant degree of cladding swelling, and (3) avoid underestimating the associated reduction in assembly flow area. The possibility of ballooning during an AOO transient or postulated accident increases as the fuel pin pressure during the event increases.
Additional Information	None.

ID	**RAC 4.2-3.4**
Acceptance Criterion	The maximum temperature of the fuel slug during postulated accidents shall be less than the melting temperature of the fuel.
Parent Requirement	RAC 4.2-3
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C.ii
Compliance Description	Establish fuel coolability criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	In severe accidents, the large and rapid deposition of energy in the fuel or insufficient cooling could result in fuel melting and relocation of molten fuel within the pin. If fuel melting coincides with failure of the cladding, it could also result in relocation of the molten fuel outside the cladding. Molten fuel relocation outside the fuel pin may adversely impact effective cooling of the core. Molten fuel relocation inside or outside the fuel pins may impact core reactivity feedback and lead to propagation of fuel failures. Design limits must be established to ensure a coolable geometry is not adversely impacted by relocation of molten fuel in to the coolant flow channel. Preventing fuel melting from occurring during postulated accidents is an adequate and conservative design limit to address this concern. If the criterion to prevent fuel melting is too restrictive, other design criteria may be used, if justified. The effects of high burnup and fast neutrons on the mechanical and thermal properties of the fuel should be considered.
Additional Information	None.

ID	**RAC 4.2-3.5**
Acceptance Criterion	Structural deformation of fuel assembly components due to the combined loads from accident conditions and natural phenomena shall not prevent the ability to adequately cool the core during postulated accidents.
Parent Requirement	RAC 4.2-3 SFR-DC 2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C.v
Compliance Description	Perform analyses to assess impact of external forces.
Compliance Specific Considerations	Establish fuel assembly component strength based on test data, analysis, engineering evaluations, or industry standards. Strengths of fuel assembly components may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components could have serious consequences, allowable values should bound a large percentage of the distribution of component strengths. Therefore, ASME Code values and procedures may be used when appropriate for determining yield and ultimate strengths. To meet the requirements of SFR-DC 2, an appropriate combination of loads from natural phenomena (e.g., earthquakes) and accident conditions must be considered to ensure that fuel system coolability can be maintained. See Appendix A for further discussion of methods for assessing structural deformation due to external forces.
Additional Information	None.

ID	**RAC 4.2-3.6**
Acceptance Criterion	Hydraulic loads, when combined with loads from natural phenomena, shall not unseat a fuel, reflector, or shield assembly and cause a reduction in coolant flow that could prevent the ability to adequately cool the fuel assembly during postulated accidents.
Parent Requirement	RAC 4.2-3 SFR-DC 2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C
Compliance Description	Perform analyses to demonstrate holddown capability
Compliance Specific Considerations	Unseating a fuel assembly may result in a significant reduction of the coolant flow to the fuel assembly. Unseating a reflector or shield assembly may result in a reduction of coolant flow to fuel assemblies. To meet the requirements of SFR-DC 2 as it relates to combining loads, an appropriate combination of hydraulic loads during postulated accidents and loads from natural phenomena should be considered. The worst-case combined loads for postulated accidents should not exceed the holddown capability of the fuel, reflector, or shield assembly. As an alternative to the above design limit, analyses can be performed to demonstrate adequate core cooling if unseating of a fuel, reflector, or shield assembly is predicted during a postulated accident.
Additional Information	None.

ID	**RAC 4.2-4 Control/Standby Rod Insertability Criteria**
Acceptance Criterion	Reactivity control assembly criteria shall be established for all severe damage mechanisms that may occur during postulated accidents to ensure that control rods can be fully inserted when required.
Parent Requirement	SFR-DC 26 10 CFR 50.62
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C
Compliance Description	Establish control rod insertability criteria based on test data, analysis, engineering evaluations, or industry standards.
Compliance Specific Considerations	SFR-DC 26 requires that the reactivity control system be designed with margin to have a capability of reliably controlling reactivity changes during postulated accidents. 10 CFR 50.62 requires an alternate means to shut down the core during ATWS events. Maintaining the ability to insert control rods during postulated accidents minimizes the potential for and extent of fuel damage, thus reducing the amount of fission products released to the primary coolant system in the event a postulated accident occurs.
Additional Information	For SFRs, control rod insertability criteria and core coolability criteria are not as strongly coupled as for LWRs due to the use of separate fuel assemblies and reactivity control assemblies for SFRs.

ID	**RAC 4.2-4.1**
Acceptance Criterion	Structural deformation of control assemblies due to the combined loads from accident conditions and natural phenomena shall not prevent the ability to insert control rods during postulated accidents.
Parent Requirement	RAC 4.2-4 SFR-DC 2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C.v
Compliance Description	Perform analyses to assess impact of external forces. Perform impact tests to characterize dynamic crush strengths of components.
Compliance Specific Considerations	Establish control assembly component strength based on test data, analysis, engineering evaluations, or industry standards. Strengths of control assembly components may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components could have serious consequences, allowable values should bound a large percentage of the distribution of component strengths. Therefore, ASME Code values and procedures may be used when appropriate for determining yield and ultimate strengths. To meet the requirements of SFR-DC 2, an appropriate combination of loads from natural phenomena (e.g., earthquakes) and accident conditions must be considered to ensure that control rod insertability can be maintained. See RAC 4.2-6, Specific Considerations Item C, for further discussion of methods for assessing structural deformation due to external forces.
Additional Information	None.

ID	**RAC 4.2-4.2**
Acceptance Criterion	Hydraulic loads, when combined with loads from natural phenomena, shall not unseat a reactivity control assembly that could prevent the complete insertion of control rods during postulated accidents.
Parent Requirement	RAC 4.2-4 SFR-DC 2
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C
Compliance Description	Perform analyses to demonstrate holddown capability
Compliance Specific Considerations	Unseating a reactivity control assembly may result in significant interference with inserting a control rod. To meet the requirements of SFR-DC 2 as it relates to combining loads, an appropriate combination of hydraulic loads during postulated accidents and loads from natural phenomena should be considered. The worst-case combined loads for postulated accidents should not exceed the holddown capability of the reactivity control assembly. As an alternative to the above design limit, analyses can be performed to demonstrate complete control rod insertion if unseating of a control assembly is predicted during a postulated accident.
Additional Information	None.

ID	**RAC 4.2-5 Fuel System Description**
Acceptance Criterion	The fuel system description and design drawings shall provide information necessary to verify that the fuel system design bases are met.
Parent Requirement	SFR-DC 10 SFR-DC 26 SFR-DC 35 10 CFR 50.62
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 2
Compliance Description	Provide documentation in SAR Section 4.2.
Compliance Specific Considerations	The fuel system description and design drawings should provide an accurate representation and supply the information needed for regulatory approval. The description should include the following fuel system information and associated tolerances: • Type and metallurgical state of the cladding • Cladding outside diameter • Cladding inside diameter • Cladding inside roughness • Slug outside diameter • Slug roughness • Slug density • Slug length • Slug alloy composition • Shield slug parameters • Fuel column length • Overall pin length • Rod internal void volume • Fill gas type and pressure • Sorbed gas composition and content (i.e., surface gas) • End plug dimensions • Wire wrapping dimensions • Fissile enrichment • Equivalent hydraulic diameter • Design-specific burnup limit • Control rod descriptions, dimensions, and lifetime limits • Fit of control rod interference with surrounding structure The description should include the following design drawings and dimensions: • Fuel assembly cross section • Fuel assembly outline • Fuel pin schematic

<table>
<tr><td>ID</td><td>RAC 4.2-5 Fuel System Description</td></tr>
<tr><td></td><td>Wire wrap location
Inlet and outlet nozzles
Control rod duct with respect to control rod dimensions
Control rod assembly cross section
Control rod assembly outline
Control rod schematic
Orifice and source assembly outline
Additional guidance related to required description of the fuel system is provide in Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants.”</td></tr>
<tr><td>Additional Information</td><td>None.</td></tr>
</table>

ID	**RAC 4.2-6 Fuel System Design Evaluation**
Acceptance Criterion	Design evaluations shall be performed using acceptable methods to demonstrate that the fuel system design bases are met during conditions of normal operation, AOOs, and postulated accidents.
Parent Requirement	SFR-DC 10 SFR-DC 26 SFR-DC 35 10 CFR 50.62
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 3
Compliance Description	Design evaluations (methods) include operating experience, prototype testing, and analytical predictions.
Compliance Specific Considerations	Design evaluations are performed to ensure compliance with SFR-DC. • Compliance with SFR-DC 10 significantly reduces the likelihood of fuel failures during normal operations or AOOs, thereby minimizing the possible release of fission products. In addition, preventing fuel damage during normal operation and AOOs may also reduce the severity of fuel damage during an accident. • Compliance with SFR-DC 26 ensures the ability to insert control rods during postulated accidents minimizes the extent of fuel damage, thus reducing the amount of fission products released to the primary coolant system in the event an accident occurs. • Compliance with SFR-DC 35 ensures that fuel system damage will not interfere with effective core cooling, thereby minimizing the potential for offsite release during postulated accidents. RCP 4.2 addresses evaluations performed related to mechanical criteria and normal operation. Other evaluations are addressed in RCP 4.3 (evaluations related to nuclear criteria and normal operation), RCP 4.4 (evaluations related to thermal hydraulic criteria and normal operation), and RCP 15 (evaluations related to AOO and postulated accidents). Design evaluations use acceptable methods with conservative treatment of uncertainties in the values of important parameters. Many of these methods will be presented generically in technical reports and will be incorporated in the SAR by reference. New fuel designs, new operating limits (e.g., pin burnup and power), and the introduction of new materials to the fuel system require an evaluation to verify that existing design-basis limits, analytical models, and evaluation methods remain applicable for the specific design for normal operation, AOOs, and postulated accidents. Design evaluation methods include operating experience, prototype testing, and analytical predictions.

ID	**RAC 4.2-6 Fuel System Design Evaluation**
	A. Operating Experience Operating experience with fuel systems of the same or similar design should be described, including the maximum burnup experience. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses are used. Design criteria for fretting wear, cladding erosion, and cladding corrosion might be addressed in this manner. B. Prototype Testing When conclusive operating experience is not available, as with the introduction of a design change, prototype testing may be used and should be described. Out-of-reactor tests should be performed, when practical, to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested before irradiation, but the following out-of-reactor tests serve as a guide for tests to consider: • Duct structural tests • Control rod structural and performance tests • Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping) • Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, fuel pin fretting, and assembly wear and life) In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be conducted. The maximum burnup or fluence experience associated with such tests should be considered in relation to the specified maximum burnup or fluence limit for the new design. The following phenomena serve as a guide for tests to consider: • Fuel and control rod growth • Fuel pin bowing • Fuel pin, wire wrap, and duct chemical and metal interactions • Fuel pin fretting • Fuel assembly growth • Duct wear, swelling and distortion • Fuel swelling (FCMI) • Clad wastage (i.e. corrosion, erosion, clad decarburization, FCCI, eutectic reaction) • Fuel pin integrity • Wire wrap relaxation • Duct wear characteristics In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished before operation of the design's full core. The inability to perform in-reactor testing may result from an incompatibility of the new

ID	**RAC 4.2-6 Fuel System Design Evaluation**
	design with the previous design. In such cases, special attention should be given to the surveillance plans. C. Analytical Predictions Some design bases and related parameters can only be evaluated with analytical methods. The analytical methods that are used to make performance predictions must be identified and described. The validity of analytical models used to predict the performance of the fuel system design, and their applicability up to the design's specified burnup and power limit, should be demonstrated for each fuel type used in the reactor. An exception may be made for prototype test assemblies, in which case only an estimate of the maximum burnup and power needs to be provided. Specific aspects of the analytical methods that should be addressed are described below. Fuel Temperatures (Stored Energy). Fuel temperatures and stored energy during normal operation serve as input to several fuel system performance calculations. Temperature calculations require complex computer codes that model many different phenomena. Important phenomenological models include the following: • Radial power distribution • Fuel and cladding temperature distribution • Burnup distribution in the fuel • Thermal conductivity of the fuel, cladding • Thermal expansion of the fuel and cladding • Fission gas production and release • Solid and gaseous fission product swelling • Fuel restructuring and relocation • Diffusion of fuel constituents • Fuel and cladding dimensional changes • Fuel-to-cladding heat transfer coefficient • Thermal conductivity of the gas mixture • Thermal conductivity in the Knudsen domain • Fuel-to-cladding contact pressure • Heat capacity of the fuel and cladding • Swelling and creep of the cladding • Rod internal gas pressure and composition • Sorption of helium and other fill gases (on surfaces) • Cladding-to-coolant heat transfer coefficient • Cladding wastage (erosion, corrosion) • FCCI

ID	**RAC 4.2-6 Fuel System Design Evaluation**
	Because of the strong interaction between these models, overall code behavior should be checked against standard problems or benchmarks and, if available, results from other analytical methods. Structural Deformation from External Forces. See Appendix A for discussion on the evaluation of structural deformation from external forces. Cladding Rupture and Flow Blockage (Ballooning). The methods for postulated accident evaluation may include cladding rupture and flow blockage models. These empirical models should be compared with relevant data. These models should account for the phase transformation in the cladding at high temperatures. Fuel Pin Pressure. The thermal performance code for calculating fuel temperatures should be used to calculate fuel pin pressures in conformance with the fuel damage criteria. This calculation should account for uncertainties in the estimated pin powers, code models, and fuel pin fabrication. Models for fission gas release from fuel to plenum region should be justified. Ensure that conservatisms incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel pin pressures. Fission Product Inventory. A description and justification should be provided for the model used to predict the release of volatile fission products from the fuel for accidents in which the fuel temperature does not exceed the temperature experienced during normal operation and AOOs. When used with nuclide yields, this model will define the inventory of volatile fission products that is available for release from the fuel pin if the cladding were breached, sometimes referred to as gap inventory. A description and justification should be provided for the model used to predict the release of fission products to the coolant from postulated accidents in which the fuel temperature exceeds the temperature experienced during normal operation and AOOs. Duct Bowing. In addition to the potential impact of mechanical loads on fuel system components, duct bowing can significantly impact the reactivity feedback during power maneuvering during normal operation and during AOOs and postulated accidents. A description and justification should be provided for the model used to predict duct bowing. Because of the strong interaction between duct bowing, coolant temperatures, and core power level, overall model behavior should be checked against standard problems or benchmarks and, if available, results from other analytical methods.
Additional Information	American Nuclear Society (ANS) 5.4 presents an approved method for analyzing release during accidents and situations that do not involve accidents in which the fuel temperature exceeds the temperature experienced during normal operation and AOOs. ANS 5.4 also provides an acceptable analytical model for calculating the release of volatile fission products from oxide fuel pellets during normal steady-state conditions.

ID	**RAC 4.2-7 Testing and Inspection of New Fuel**
Acceptance Criterion	Testing and inspection shall be performed for new fuel to ensure that the fuel is fabricated in accordance with the design basis and that it reaches the plant site and is loaded in the core without damage.
Parent Requirement	SFR-DC 10
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 4.A
Compliance Description	Develop testing and inspection program
Compliance Specific Considerations	Testing and inspection plans for new fuel should verify cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Quality control reports should document the details of the manufacturer's testing and inspection programs and should be referenced and summarized in the safety analysis report. The program for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described in the safety analysis report.
Additional Information	None.

ID	**RAC 4.2-8 Online Fuel System Monitoring**
Acceptance Criterion	Online methods or surveillance programs shall be developed to detect fuel pin failure or reactivity control/ absorber pin failure.
Parent Requirement	SFR-DC 10 SFR-DC 26 10 CFR 50.62
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 4.B
Compliance Description	Design fuel monitoring system and develop surveillance program.
Compliance Specific Considerations	Online fuel failure monitoring should be performed to detect anomalies. Both the sensitivity of the instruments should be defined and the program to use the instruments should be developed. NUREG-0401 and NUREG/CR-1380 describe common detection methods that are acceptable. For metallic fuel, failed fuel detection is not a crucial safety function due to compatibility of the metal alloys with the sodium coolant. Surveillance is also needed to ensure that control rods are not losing reactivity. Periodic reactivity worth tests should be considered in developing the surveillance program.
Additional Information	None.

ID	**RAC 4.2-9 Post Irradiation Surveillance**
Acceptance Criterion	A post-irradiation examination and surveillance program to detect anomalies or confirm expected performance shall be established for each fuel and reactivity control assembly design.
Parent Requirement	SFR-DC 10 SFR-DC 26 SFR-DC 35 10 CFR 50.62
Basis	NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 4.C
Compliance Description	Develop fuel assembly and reactivity control assembly surveillance programs.
Compliance Specific Considerations	The extent of an acceptable post-irradiation examination and surveillance program will depend on the history of the fuel design being considered (i.e., whether the proposed fuel design is the same as current operating fuel or incorporates new design features). For a fuel design similar to that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel pin failure, pin bowing, dimension changes, or crud deposition. The program should also commit to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should address the disposition of failed fuel. In addition to the plant-specific surveillance program, a continuing fuel surveillance effort should exist for a given type, make, or class of fuel that can be suitably referenced by all plants using similar fuel. In the absence of such a generic program, the reviewer should expect more detail in the plant-specific program. For a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes should be described. This program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with the prototype testing discussed in the design evaluation for prototype testing. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.
Additional Information	There are two types of the surveillance program. One is at a special pit or pool in the plant, another is at a hot cell facility. The former is typically for limited non-destructive tests for a core assembly that may be reloaded for further cycles. The latter is more suitable for comprehensive destructive tests. Both surveillance programs require extensive efforts to develop surveillance equipment and inspection procedures with safety analysis on the testing facility and core components to be inspected. In addition, these programs should not impact plant operation schedules and safety.

4. REFERENCES

1. [NOTE: *Add reference for plant specific SAR requirements or reference appropriate NUREG document.*]

2. [NOTE: *Add reference for plant specific regulatory requirements or reference 10 CFR 50 or 10 CFR 52 as appropriate.*]

3. NRC Regulatory Guide 1.232 Revision 0, “Guidance for Developing Principal Design Criteria for Non-Light Water Reactors”.

4. 10 CFR 100, “Reactor Site Criteria.”

5. 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.”

6. American National Standards Institute, ANSI/ANS 5.4, “Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel.”

7. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Facility Components.”

8. NRC Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants.”

9. NUREG-0401, “Fuel Failure Detection in Operating Reactors.”

10. NUREG-0800, Section 4.2, “Fuel System Design.”

11. NUREG/CR-1380, “Assessment of Current Onsite Inspection Techniques for Light-Water Reactor Fuel Systems.”

APPENDIX A - SRP REGULATORY REQUIREMENTS OR ACCEPTANCE CRITERIA NOT APPLICABLE FOR SFR

1. 10 CFR 50.46, 10 CFR 50.34, and 10 CFR 50.67 – These requirements relate to acceptance criteria for LOCA and ECCS performance. These requirements are very LWR technology specific. All postulated accidents, including LOCA, must meet the core coolability requirement specified in 10 CFR 50 Appendix A GDC 35. 10 CFR 50.46 identifies core coolability requirements necessary for meeting GDC 35 requirements during an LWR LOCA. These LWR specific requirements are neither appropriate nor necessary for an SFR LOCA. Core coolability requirements for SFR postulated accidents (including LOCA) are required by PDC 35 with appropriate event specific acceptance criteria defined as necessary.

2. 10 CFR 50.67 – This requirement applies to operating licenses issued prior to January 10, 1997.

3. 10 CFR 52.47(b)(1) and 10 CFR 52.80(a) – Licensing requirements are based on the 10 CFR 50 regulatory process. [*Note: Revise statement if 10 CFR 52 regulatory process is used.*]

4. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.iv – For an SFR with metallic fuel, oxidation and hydriding are not a concern. However, corrosion and erosion are concerns. Crud is a phenomenon of concern for LWR fuel, but not for SFR fuel.

5. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.v – The cladding irradiation effects of concern for SFRs are creep and swelling, while for LWRs it is growth. Shadow corrosion is an LWR concern, not an SFR concern.

6. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.A.vi – Reorientation of hydrides in the cladding is not a phenomena of concern for SFR fuel types.

7. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.B.i – Hydriding is an LWR fuel phenomena and does not occur in fuel types for sodium fast reactors.

8. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.B.ii – Cladding collapse is an LWR fuel phenomena and does not occur for SFR fuel.

9. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.B.iii – Traditional practice for LWRs assumes that failures will not occur due to overheating of the cladding if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. The thermal margin criteria are measures of margin to the degradation in heat removal capability due to departure from nuclear boiling for PWRs or dryout (steam blanketing) for BWRs. These phenomena are not expected to occur in an SFR during normal operation, AOOs, or postulated accidents; therefore, these phenomena are not limiting for SFRs.

10. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 1.C.i – Cladding embrittlement is a phenomena of LWRs due to high temperatures, oxidation, and hydriding during a LOCA. Coolant inventory is lost followed by the injection of relatively cold coolant (e.g., ECCS), which cause embrittlement-induced failure due to rapid quenching.

11. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 3.C.i – Since LOCA is not a major concern for an SFR, Appendix K does not apply for an SFR.

12. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 3.C.ii – Densification effects are not a major concern for SFRs, especially for metallic fuel.

13. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 3.C.iii – The methodology discussed for the prediction of rod bowing is applicable for LWRs and not for SFRs.

14. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 3.C.iv – Cladding collapse is not a mechanism of concern for SFRs because the fuel rod internal pressure is expected to always exceed that of the coolant.

15. NUREG-0800, Section 4.2, Part II, SRP Acceptance Criterion 3.C.viii – Metal/water reaction is a phenomena associated with LWR fuel at high fuel temperature during a LOCA.

16. Appendix B of NUREG-0800, Section 4.2 – The appendix provides interim acceptance criteria and guidance for reactivity-initiated accidents for LWRs. This material is not applicable for SFRs.

APPENDIX B - EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE TO EXTERNALLY APPLIED FORCES

I. BACKGROUND

Earthquakes and postulated accidents in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. This appendix describes the review that may be performed of the fuel assembly structural response to seismic and postulated accident loads. NUREG0609, NUREG/CR-1018, NUREG/CR-1019, and NUREG/CR-1020 provide background or reference material for this appendix. For SFRs, they are traditionally grouped in three accident categories: The reactivity-initiated accident (RIA) involves a sudden and rapid insertion of positive reactivity such as a control element/rod withdrawal (CRW) and subsequent increase in core power. Loss of flow (LOF) accident imply inability to provide adequate forced convection flow to cool the core due to primary pump failures. Loss of heat sink (LOHS) accident involves failures in heat removal paths that are relied on during normal operation. During such events, the fuel and coolant temperatures increase, prompting the action of the engineered safety systems and/or reactor's inherent response. There are multiple reactivity feedback mechanisms that work in tandem in SFRs to lower the reactor power during transients in response to an uncontrolled increase in core and primary coolant temperatures. These mechanisms include the feedback due to Doppler broadening of neutron cross-sections, changes in primary sodium coolant density, fuel axial expansion, core radial expansion, and control element/rod driveline extension.

II. ANALYSIS OF LOADS

1. Input

   Input for the fuel assembly structural analysis comes from the results of the primary coolant system and reactor internals structural analysis, which is reviewed by the organization responsible for the review of mechanical engineering issues. Input for the fuel assembly response to a seismic should include (1) motions of the core plate, core shroud, fuel alignment plate, or other relevant structures (these motions should correspond to the break that produced the peak fuel assembly loadings in the primary coolant system and reactor internals analysis) and (2) transient reactivity insertions due to core compactions that increase temperature to the fuel assembly. If the earthquake loads are large enough to produce a nonlinear fuel assembly response, input for the seismic analysis should use structure motions corresponding to the reactor primary coolant system analysis for the safe-shutdown earthquake (SSE). If a linear response is produced, a spectral analysis may be used in accordance with the guidelines of RG 1.60.

2. Methods

   Analytical methods used in performing structural response analyses should be reviewed. The appropriateness of numerical solution techniques should be justified. Linear and nonlinear structural representations (i.e., the modeling) should also be reviewed. Experimental verification of the analytical representation of the fuel assembly components should be provided when practical. The applicant should work a sample problem of a simplified nature, which the reviewer will compare with either hand calculations or results generated with an independent code (NUREG/CR-1019). Although the sample problem should use a structural representation that is as close as possible to the design in question (and, therefore, would vary from one vendor to another), the applicant may make simplifying assumptions (e.g., one might use a three-assembly core region with continuous sinusoidal input). The sample problem should be designed to

exercise various features of the code and reveal their behavior. The sample problem comparison is not, however, designed to show that one code is more conservative than another, but rather to alert the reviewer to major discrepancies so that an explanation can be sought.

3. Uncertainty Allowances

The fuel assembly structural models and analytical methods are likely to be conservative; input parameters are also conservative. However, to ensure that the fuel assembly analysis does not introduce any non-conservatisms, two precautions should be taken—(1) if it is not explicitly evaluated, impact loads from the seismic analysis should be increased by about XX percent to account for a reactivity insertion, which is associated with core compactions and (2) conservative margin should be added if any part of the analysis exhibits pronounced sensitivity to input variations. Variations in resultant loads should be determined for positive variations in input amplitude and frequency of 10 percent; variations in amplitude and frequency should be made separately, not simultaneously. A factor should be developed for resultant load magnitude variations of more than 15 percent. For example, if +10-percent variations in input magnitude or frequency produce a maximum resultant increase of 35 percent, the sensitivity factor would be 1.2. Since resonances and pronounced sensitivities may be plant dependent, the sensitivity analysis should be performed on a plant-by-plant basis until the reviewer is confident that further sensitivity analyses are unnecessary, or it is otherwise demonstrated that the analyses performed are bounding.

4. Audit

The reviewer should perform independent audit calculations for a typical full-sized core to verify that the overall structural representation is adequate. An independent audit code (NUREG/CR-1019) should be used for this audit during the generic review of the analytical methods.

5. Combination of Loads

To meet the requirements of GDC 2, as it relates to combining loads, an appropriate combination of loads from natural phenomena and accident conditions must be made. Loads on fuel assembly components should be calculated for each input (i.e., seismic and limiting accident) as described in Subsection II.1 of this appendix, and the resulting loads should be added by the square-root-of-sum-of-squares method. These combined loads should be compared with the component strengths described in Subsection III according to the acceptance criteria in Subsection IV.

III. DETERMINATION OF STRENGTH

1. Ducts

All modes of loading (e.g., flat-to-flat and corner-to-corner loadings) should be considered, and the vendor's laboratory duct strength tests should represent the most damaging mode. Test procedures and results should be reviewed to assure that the appropriate failure mode is being predicted. The review should also confirm that (1) the testing impact velocities correspond to expected fuel assembly velocities and (2) the crushing load P(crit) has been suitably selected from the load-versus-deflection curves. Because of the potential for different test rigs to introduce measurement variations, the review of the test procedure will evaluate the duct strength test equipment. The consequences of duct deformation is small. Gross deformation of ducts in an SFR control assembly would be needed to interfere with control rod insertion during an SSE. Core compactions due to gross deformations of the fuel assembly ducts would result in significant increases in peak cladding temperature. Therefore, average values are appropriate, and the

allowable crushing load P(crit) should be the 95- percent confidence level on the true mean as taken from the distribution of measurements on unirradiated production ducts at (or corrected to) operating temperature. While P(crit) will increase with irradiation, ductility will be reduced. The extra margin in P(crit) for irradiated ducts is thus assumed to offset the unknown deformation behavior of irradiated ducts beyond P(crit).

2. Components Other Than Ducts

Strengths of fuel assembly components other than ducts may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components (e.g., fracturing of strip rails, locking plate, or fragmentation of fuel pins) could be more serious than duct deformation, allowable values should bound a large percentage (about 95 percent) of the distribution of component strengths. Therefore, ASME Code values and procedures may be used when appropriate for determining yield and ultimate strengths. Specification of allowable values may follow the ASME Code requirements and should consider buckling and fatigue effects.

IV. ACCEPTANCE CRITERIA

Two principal criteria apply for the combined natural phenomena with accidents (1) fuel pin fragmentation must not occur as a direct result of the combined loads and (2) fuel temperature limits must not be exceeded. The first criterion is satisfied if the combined loads on the fuel pins and components other than ducts remain below the allowable values defined above. The second criterion is satisfied by a coolable geometry analysis. If combined grid loads exceed P(crit), then duct deformation must be assumed and the coolable geometry analysis must include the effects of distorted fuel assemblies. An assumption of maximum credible deformation (i.e., fully collapsed ducts onto pin bundle) may be made unless other assumptions are justified. (3) Control rod insertability is a third criterion that must be satisfied. Power distributions from the worst-case accident that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load.

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