TerraPower

November 11, 2020

TP-LIC-LET-2020-1102
Project Number 99902087

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

Subject: TerraPower®, LLC – Advanced SFR Fuel Assembly Qualification Plan

References: 1. Letter – TerraPower LLC to Document Control Desk, "*Regulatory Guidance Development Report*," July 16, 2020 (ML20209A155)

TerraPower, LLC received a Regulatory Assistance Grant from the DOE to develop an Advanced Fuel Qualification Methodology Report. The Report will provide regulatory guidance, describe methodologies, and identify regulatory and qualification criteria for Sodium Fast Reactors (SFR) metallic fuel. The final report will include a Regulatory Guidance Development Report, an Advanced SFR Fuel Assembly Qualification Plan, a Fuel Pin Type 1 Qualification Plan, and a Fuel Pin Type 1B Qualification Plan. The Regulatory Guidance Development Report was submitted to the NRC for review and feedback on July 16, 2020, TerraPower LLC submitted (see Ref. 1).

The Advanced SFR Fuel Assembly Qualification Plan is the second submittal of the Advanced Fuel Qualification Methodology Report. The objective of the fuel assembly qualification plan is to confirm all aspects of the fuel assembly design and fabrication process will provide reliable and safe operation of an advanced Sodium-cooled Fast Reactor (SFR) where advanced metallic fuel pins will be used. The report provides a process to establish the fuel assembly qualification plan.

Portions of the Advanced SFR Fuel Assembly Qualification Plan are considered proprietary, and TerraPower requests it be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390. Additionally, the information indicated as proprietary has also been determined to contain Export Controlled Information. This information must be protected from disclosure pursuant to the requirements of 10 CFR 810. Enclosures 1 and 2 to this report provide the approved proprietary and non-proprietary versions of this report, designated as AFQMG-ENG-PLAN-0002 and AFQMG-ENG-PLAN-0002R, respectively. An affidavit supporting the withholding request is provided in

Enclosure 3. TerraPower authorizes the NRC to reproduce and distribute the submitted non-proprietary content, as necessary, to support the conduct of their regulatory responsibilities.

This submittal is a White Paper. TerraPower is requesting the NRC review and evaluate the Advanced SFR Fuel Assembly Qualification Plan and provide preliminary feedback on the approach. TerraPower understands the NRC will perform a preliminary assessment to understand the scope and content. After the preliminary assessment, NRC may provide a bulletized series of questions and comments to address the observations.

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

Enclosures:
1) AFQMG-ENG-PLAN-0002, Advanced SFR Fuel Assembly Qualification Plan, Revision 0 (Proprietary)
2) AFQMG-ENG-PLAN-0002R Advanced SFR Fuel Assembly Qualification Plan, Revision 0 (Non-Proprietary)
3) Affidavit Supporting Request for Withholding from Public Disclosure (10 CFR 2.390)

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ENCLOSURE 2

Advanced SFR Fuel Assembly Qualification Plan, Revision 0

(Non-Proprietary)

TerraPower™

ADVANCED SFR FUEL ASSEMBLY QUALIFICATION PLAN

Non-Proprietary

October 2020

15800 Northup Way
Bellevue, WA 98008

The proprietary information is redacted in this document, and is denoted as trade secrets (TS)

REVISION HISTORY

Revision No.	Effective Date	Affected Section(s)	Description of Change(s)
0	10/27/2020	All	Initial Release

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1 INTRODUCTION

The objective of the fuel assembly qualification plan is to confirm that all aspects of the fuel assembly design and fabrication process will provide reliable and safe operation of an advanced Sodium-cooled Fast Reactor (SFR) where advanced metallic fuel pins will be used. This plan satisfies the relevant design criteria in Appendix B "Guidance for Developing Principal Design Criteria for Non-Light-Water-Reactors" of RG 1.232 [1] as well as the Regulatory Acceptance Criteria described in Section 4.1. This report provides a process to establish the fuel assembly qualification plan with the following content:

- Background for Fuel Assembly Design, Core Restraint System, and Core Seismic Analysis
- Applicable Regulations and Standards
- Fuel Assembly Design Criteria Associated with Regulatory Acceptance Criteria (RAC)
- Core Mechanical Computer Code Descriptions
- Fuel Assembly Design Evaluation by Phenomena Identification and Ranking Table (PIRT)
- Major Tasks for Fuel Assembly Qualification

The scope of this plan includes not only the fuel assembly design, but also the core mechanical behaviors related to the interactions between fuel assemblies, control assemblies, shielding or reflector assemblies. Note the fuel pin qualification plan is to be provided in AFQM-ENG-PLAN-0001 [2].

This report is a deliverable of the Advanced Fuel Qualification Methodology project (DOE-FOA-0001817 Regulatory Assistance Award).

2 BACKGROUND

2.1 Fuel Assembly Design Descriptions

The Travelling Wave Reactor (TWR) uses a typical SFR fuel assembly design that has a hexagonal cross-section and is comprised of a handling socket with top load pads, a duct tube with above core load pads, fuel pin bundle/rails, upper and lower shielding blocks (or rods), orifice plates, and an inlet nozzle, as shown in Figure 2-1. The fuel pins are wrapped with wire to maintain spacing for coolant flow and to provide adequate restraint against flow-induced vibration.

The assembly design has standard requirements to ensure reactor core safety during operations and accidents. Therefore, the fuel assembly qualification plan in this document will cover various design features, either existing or advanced, that comply with the Nuclear Regulatory Commission (NRC) Standard Review Plan [3].

The detailed design features of the advanced fuel assembly will be determined by the advanced reactor developer. The baseline fuel design and bounding operational parameters are given in Table 2-1 and Table 2-2.

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Figure 2-1: Schematics of a TWR Driver Fuel Assembly

Table 2-1: Examples of Advanced Fuel Assembly Design Parameters

Parameter	Value	Unit
Cladding Outer Diameter	TS	mm
Cladding Thickness		mm
Fuel Length		m
Plenum Length		m
Fuel Type		-
Weight Fraction Zr		-
Fuel Smear Density Fraction		-
Fuel Type		
Bond		-
Cladding Material		-
Pins per Assembly		-

Table 2-2: Advanced Fuel System Bounding Operational Parameters

Parameter	Baseline Design	Relevant Ranges	Unit
Peak Burnup		TS	%FIMA
Peak Burnup			GWd/MT[1]
Average Burnup			%FIMA
Average Burnup			GWd/MT[1]
Peak DPA			DPA
Residence			EFPD
Residence			Years
Peak Linear Heat Rate			kW/m
Peak Plutonium Content			wt%

[1] Calculated using 200 MeV/fission, 238 g/mol

2.2 Core Restraint System (CRS)

During reactor operation, core assemblies are subject to bowing deformations due to the combined effects of the following phenomena:

- Thermal and fluence gradients
- Thermal and irradiation creep
- Interactions of the core restraint system

Since core assembly bowing can cause significant changes in reactivity of the core during startup, overpower, and loss-of-flow without scram transients, the assembly deformation due to the above phenomena must be controllable and predictable for safe, reliable, and economic reactor operation.

The CRS consists of core former rings connected to the Core Support Structure (CSS), a Top Load Pad (TLP) and Above Core Load Pad (ACLP) on the core assemblies, an inlet nozzle at the bottom of the core assembly, and the receptacle at the bottom of the CSS, as shown in Figure 2-2. The deflection (bow) of core assemblies is limited by the TLP or ACLP ring for a typical limited free bow CRS design. The CRS design has many competing requirements. For example, tighter gaps between load pads may reduce reactivity insertions, but may lead to core assembly handling challenges. On the other hand, larger gaps at the TLP ring may induce a higher negative power coefficient over the entire operating range including startup, but it may lead to poor seismic behavior. Therefore, optimizing dimensions of the CRS and the clearances are very important in order to meet the competing design criteria.

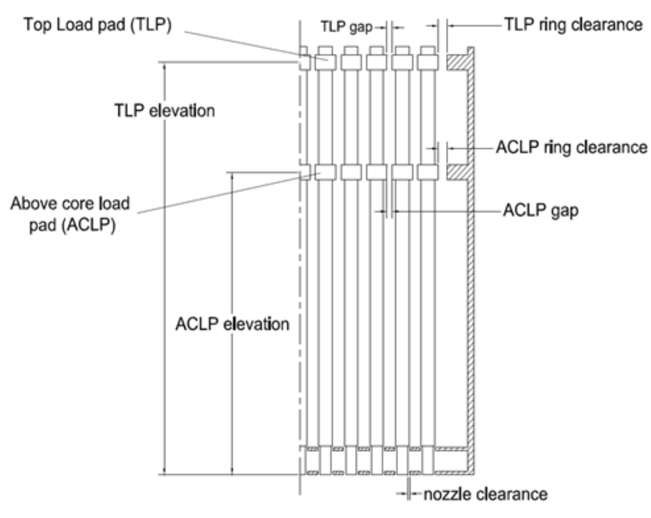

Figure 2-2: Schematics of a Typical Limited Free Bow CRS

The CRS design is complex due to these competing parameters and nonlinearities introduced by inter-assembly gaps and material effects at high temperature and neutron fluence. As such, a computer code to simulate core component behavior should be developed and Verified and Validated (V&V'ed) by benchmarking against out-of-pile and in-pile test results (see Section 7.5 for more details). However, it is challenging to get in-pile test results until a reactor starts operation; as such, it is recommended to perform simulated out-of-pile testing by using beginning-of-life (BOL) and end-of-life (EOL) projected geometries (dilated and twisted) with a range of clearances to mimic operating conditions. In addition, comparison to an existing computer code would be a useful verification method. More background information for the V&V of the CRS design code for liquid metal fast breeder reactors (LMFBR) can be found in the International Working Group on Fast Reactors (IWGFR) Co-ordinated Research Programme (CRP) organized by the International Atomic Energy Agency (IAEA) [4] and [5].

2.3 Core Seismic Analysis

Another important aspect to the evaluation of core assembly design is structural integrity under seismic loads. It is a multi-step analysis consisting of structure-soil-interaction (SSI), reactor equipment system, reactor core, and single core component analyses. The reactor core analysis can be performed using a single row or full core model as shown in Figure 2-3.

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Figure 2-3: A Typical Reactor Seismic Analysis Outline

The core assembly detailed model evaluates structural integrity of small components in detail by using limiting responses obtained from the reactor core analysis (see Section 0 for details). Both the reactor core and single core component models shall be V&V'ed by testing (see Section 7.2. for details). IAEA efforts on the LMFBR seismic analysis can be found in [6], [7], and [8], which can be used for additional V&V.

3 APPLICABLE REGULATIONS AND STANDARDS

In addition to the NRC SRP 4.2 [3], the following regulations and standards have been identified to be applicable to the fuel assembly qualification plan.

3.1 NRC RG 1.232 Guidance for Developing Principal Design Criteria (DC) for Non-Light-Water Reactors

3.1.1 SFR-DC 2: Design Bases for Protection against Natural Phenomena

"Structures, Systems, and Components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed."

- This criterion shall apply to reactor core seismic analysis that evaluates the structural integrity of fuel assemblies and their components. In addition, reactivity insertion effects due to core compactions or control rod displacements can be evaluated as well as control rod insertability and scram time under seismic loads.

3.1.2 SFR-DC 10: Reactor Design

"The reactor core and associated coolant, control, and protection systems shall be designed with an appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

- This criterion shall apply to analysis of the core restraint system that evaluates reactivity insertions due to fuel assembly deformations such as bow, tilt, twist, or swelling induced by high irradiation and/or thermal flux environment.
- This criterion shall apply to analysis of core assembly duct dilation that is a function of pressure differential, fast neutron flux, temperature, and time. Excessive dilation could cause handling failures when attempting core assembly withdrawal or insertion operations due to excessive contact forces between neighboring ducts or insufficient clearance at the load planes.
- This criterion shall apply to structural analysis of fuel assemblies. This includes load cases such as shipping/handling, assembly drop scenarios, and high temperature fatigue failure due to power fluctuations and core heat-up/cool-down cycles.

3.1.3 SFR-DC 35: Emergency Core Cooling

"A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the reactor core during and following postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented."

"Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure."

- This criterion shall be used to evaluate if fuel assemblies maintain their coolable geometry under severe events that may cause non-negligible plastic deformation of components.

3.2 ASME BPV Section III, Division 5

In a letter dated August 16, 2018, the NRC established a plan to endorse the 2017 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 5, "High Temperature Reactors." There are various activities being undertaken and on-going to supplement Division 5. The most recent edition is 2019 and the NRC's draft regulatory guide will be issued by Spring 2021.

Division 5 of Section III of the BPVC provides construction rules that are applicable to an SFR. These rules are for components exceeding the temperature ranges in Division 1 and are meant for components experiencing temperatures that are equal to, or higher than, 700°F (370°C) for ferritic materials or 800°F (425°C) for austenitic stainless steels or high nickel alloys. Therefore, most of the core assembly component and core support structure design should be based on Subpart B Elevated Temperature Service.

3.3 ANS 54.1 Nuclear Safety Criteria and Design Process for Liquid-Sodium-Cooled-Reactor Nuclear Power Plants

This standard defines safety objectives, General Design Criteria (GDC), selection of Licensing Basis Events (LBEs), and Classification of Systems, Structures, and Components (SSCs) for the SFR.

4 FUEL ASSEMBLY DESIGN CRITERIA

4.1 Fuel Assembly Service Conditions

In addition to normal operating conditions, fuel assemblies shall be designed to satisfy the SFR Design Criteria (SFR-DC) 2, 10, and 35 [1] regarding the following Licensing Basis Events (LBE) defined in Table 3-1 of NEI 18-04 [9].

- Anticipated Operational Occurrences (AOOs, ≥ 1×10^{-2}/plant-year)
- Design Basis Events (DBEs, < 1×10^{-2}/plant-year and ≥ 1×10^{-4}/plant-year)
- Beyond Design Basis Events (BDBEs, < 1×10^{-4}/plant-year and ≥ 5×10^{-7}/plant-year)

4.2 Regulatory Acceptance Criteria (RAC)

The NRC's Standard Review Plan Chapter 4.2 provides detailed acceptance criteria for fuel systems to demonstrate Design Criteria (DC) will be satisfied. However, the current SRP was developed for light water reactors only. Therefore, TerraPower developed a Generic Regulatory Compliance Plan and Regulatory Acceptance Criteria for SFR fuel systems [10]. The following RACs are either directly related to the fuel assembly or affecting fuel pin or control rod criteria.

The fuel assembly RACs can be categorized as follows:

- RAC 4.2-1: Fuel Assembly Damage Criteria
    - Structural Integrity: Limits for Stress / Strain / Loading
    - Endurance and Lifetime: Limits for Creep / Fatigue / Fretting Wear / Erosion & Corrosion
    - Dimensional Stability: Limit for Dimensional Changes / Hold-down / Buckling
- RAC 4.2-2: Fuel Pin Failure Criteria (merged into 4.2-6)
    - Mechanical Fracture due to Externally Applied Forces
- RAC 4.2-3: Fuel Coolability Criteria
    - Structural Deformation due to Combined Loads
    - Fuel Assembly Lift-off
- RAC 4.2-4: Control Rod Insertability Criteria (induced by neighboring fuel assemblies)
    - Structural Deformation due to Combined Loads
    - Control/Standby Assembly Lift-off
- RAC 4.2-5: Fuel System Description (see Section 2.1)
- RAC 4.2-6: Fuel System Design Evaluation (see Section 6)
- RAC 4.2-7: Testing and Inspection of New Fuel (see Section 7.5.1)
- RAC 4.2-8: Online Fuel System Monitoring (see Section 7.5.2)
- RAC 4.2-9: Post Irradiation Surveillance (see Section 7.5.3)

4.3 Design Basis Criteria and Supporting Information

Tables 4-1 through 4-4 describe the design basis criteria, available supporting data, and relevant testing or analysis activities required for each of the RACs for fuel assemblies. Note only the subset of RACs related to the fuel assembly are included in the tables.

Table 4-1: Design Basis Criteria and Supporting Information to Prevent Fuel Assembly Damage

Relevant RAC	Acceptance Criterion	Applicable Design Basis Criteria	Available Supporting Data	Relevant Testing/Analysis Activities
4.2-1.1	Stress, strain, or loading limits for all fuel system components under normal operation and AOOs shall be established.	• ASME Section III Div. 5 • Reactor Development and Technology (RDT) Standards RDT F9-7, Appendix A of F9-8, F9-9	TS	
4.2-1.2	The cumulative number of strain fatigue cycles on all fuel system components shall be significantly less than the design fatigue lifetime.	• ASME Section III Div. 5 • RDT F9-7, F9-8 (Appendix A), F9-9		
4.2-1.3	Limits on fretting wear at contact points on all fuel system components shall be established.	• (To be determined)		
4.2-1.4	Limits on erosion and the buildup of corrosion products shall be established for all fuel system components.	• (To be determined)		

Relevant RAC	Acceptance Criterion	Applicable Design Basis Criteria	Available Supporting Data	Relevant Testing/Analysis Activities
4.2-1.6	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.	• RDT F9-7, F9-8 (Appendix A), F9-9		TS
4.2-1.9	The worst-case hydraulic loads for normal operation and AOOs shall not exceed the hold-down capability of a fuel, reflector, or shield assembly.	• (To be determined)		

* NUBOW (FFTF), BAMBOO & MARSE (Japan), OXBOW & VirDenT (TerraPower): These computer codes are used to evaluate the CRS performance by simulating core assembly behavior during operation.

Table 4-2: Design Basis Criteria and Supporting Information for Fuel Coolability under Combined Loads from Accident Conditions and Natural Phenomena

Specific RAC	Acceptance Criterion	Applicable Design Basis Criteria	Available Supporting Data	Relevant Testing/Analysis Activities
4.2-3.5	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.	• RDT F9-7, F9-8 (Appendix A), F9-9		TS
4.2-3.6	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.			

Table 4-3: Design Basis Criteria and Supporting Information for Operating Basis Earthquake (OBE)

Relevant RAC	Acceptance Criterion	Applicable Design Basis Criteria	Available Supporting Data	Relevant Testing/Analysis Activities
4.2-1.1	OBE should not require changes in the operating procedures following the earthquakes	• GEFR00728 Post-OBE Operability • RDT F 9-2T		TS
4.2-1.6	Residual changes in the core geometry should be limited to preclude core undercooling			
4.2-1.6	Residual geometry changes should not require changes in the refueling procedures			
4.2-1.2	Damage during OBE should not exceed the accepted damage limits			

Table 4-4: Design Basis Criteria and Supporting Information for Design Basis Earthquake

Specific RAC	Acceptance Criterion	Applicable Design Basis Criteria	Available Supporting Data	Relevant Testing/Analysis Activities
N/A	Reactivity insertion due to Safe Shutdown Earthquake (SSE) displacements (horizontal and vertical) should be limited to specified values	• GEFR00728 Pre-scram displacements		TS
N/A	Inelastic deformations should not significantly reduce the diametral clearances between the control rod and its guide duct	• GEFR00728 Scram-ability		
N/A	Residual changes in the core geometry should be limited to preclude core undercooling	• GEFR00728 Core Coolability		

5 CORE MECHANICAL ANALYSIS COMPUTER CODE REQUIREMENTS AND CONSIDERATIONS

A fuel assembly consists of a plurality of fuel pins and structural components such as the duct tube, nozzles, and small components. Other types of core assemblies consist of different internals with similar structural components. In addition, a reactor core consists of hundreds of core assemblies such as fuel, control/shutdown, shield, and reflector assemblies. Throughout the core lifetime, there are significant interactions between these assemblies due to dimensional changes such as bow, twist, and dilation. These dimensional changes can be induced by pressure differential, non-uniform temperature and neutron fluence distributions, or dynamic responses due to drop, handling, or seismic loads. These phenomena shall be identified and evaluated for the anticipated lifetime of the core assemblies. Therefore, the use of numerical models and/or computer codes is crucial to demonstrate all the fuel assembly design criteria can be satisfied under normal operations, AOOs, and DBEs.

5.1 Core Restraint System Models and Methods

A core restraint system shall maintain adequate dimensional stability of the core assemblies during normal operation and facilitate core assembly handling operations during reactor shutdown. Typically, the core restraint system consists of core former rings, core grid plates, core receptacles, and load pads on the core assemblies. The models and methods shall be adequate to demonstrate the following requirements:

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5.2 Pin Bundle to Duct Interaction Models and Methods

The fuel pins are spaced from each other by helically wound wires wrapped over each pin's cladding to provide proper coolant flow. Both the pin bundle and the duct are subject to dilation and distortion in both the radial and axial directions due to differential pressure, thermal expansion, thermal creep, void swelling, and irradiation creep. Excessive pin bundle tightness or looseness may occur if the gaps are sized incorrectly. As such, the numerical models/methods shall be able to simulate the fuel pin behavior related to the gaps to optimize the following aspects:

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5.3 Fuel Assembly Drop Analysis Models and Methods

Core assemblies may experience drop accidents during handling, induced by a failure or malfunction of handling machines or an interaction with neighboring core assemblies. The models and methods shall be able to evaluate the structural integrity of the core assembly and core support structures during the following drop scenarios in the reactor core:

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5.4 Core Seismic Analysis Models and Methods

As a part of the series of the reactor seismic analysis described in Section 2.3, a typical core seismic analysis flow chart is shown in Figure 5-1. Following the reactor core seismic analysis, a detailed core assembly seismic analysis should be performed as shown in Figure 5-2. The seismic models and methods shall be able to perform both the core and core assembly seismic analyses. It is noted that the time-history core seismic analysis can be performed using single row core models since a full core model may be too computationally expensive or unable to converge. The single row core model analysis must be based on sufficient conservatisms. In addition, a series of seismic tests must be performed to validate the model and method.

The reactor core shall satisfy the following requirements described in Table 4-3 and Table 4-4 to comply with NUREG-800 Standard Review Plan Chapter 4.2.

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Therefore, the model and method shall be able to evaluate the following responses.

- Maximum impact forces on core assemblies:
  - The maximum force shall be less than the critical buckling load of the duct tube and load pads.
- Maximum deformation of core assemblies
  - The maximum deformation shall be less than the limit to maintain the control rod insertability.
  - The maximum deformation shall not cause significant coolant flow area reduction in order to maintain a coolable geometry of the core.
- Maximum reactivity insertion in the core (OBE only)
  - Residual changes in the core geometry shall be limited to preclude significant changes in the core reactivity that can cause unwarranted scram.
- Control rod drop time during seismic
  - The reduction in the scram time due to deformations or impacts resulting from control rod and driveline inertial effects shall be within the prescribed limits.

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Figure 5-1 Typical Reactor Core Seismic Analysis Flow Chart

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Figure 5-2 Flow Chart for Core Assembly Seismic Analysis

6 FUEL ASSEMBLY DESIGN EVALUATION

Fuel system design evaluations are required as part of the Safety Analysis Reports (SARs) using acceptable methods to demonstrate that the fuel design basis criteria are met. Acceptable design evaluation methods include operating experience, prototype testing, and analytical predictions. Design evaluations must treat uncertainties in the values of important parameters in a conservative manner and must consider the physically feasible combinations of chemical, thermal, irradiation, mechanical, and hydraulic interactions. The evaluation of these interactions should include the effects of normal operations and design basis events, including AOOs, anticipated transients without scram (ATWS), and postulated accidents.

The high impact phenomena affecting fuel system response under design basis events were identified by performing a Phenomena Identification and Ranking Table (PIRT) assessment so that the qualification plan can be established effectively and efficiently.

Table 6-1: Importance Ranking Definitions

Importance Rank	Definition
Low (L)	Small influence on demonstrating compliance ± 1 σ variation of parameter/phenomenon has minimal impact on prediction of design criterion
Medium (M)	Moderate influence on demonstrating compliance ± 1 σ variation of parameter/phenomenon has moderate impact on prediction of design criterion
High (H)	Significant influence on demonstrating compliance ± 1 σ variation of parameter/phenomenon has significant impact on prediction of design criterion

Table 6-2: Knowledge Level Definitions

Knowledge Level	Definition
Known (K)	Approximately 70-100% of complete knowledge and understanding
Partial Known (P)	30-70% of complete knowledge and understanding
Unknown (U)	0-30% of complete knowledge and understanding

Table 6-3: PIRT for Preventing Fuel System Damage under Normal Operation and AOOs

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Stress, Strain, Loading Limit	Impact loads due to handling drop accidents				TS
Stress, Strain, Loading Limit	Withdrawal/insertion forces				

Category	**Phenomena**	**Key Influencing Parameters**	**Importance Ranking**	**Knowledge Level**	**Additional Comments**
Cumulative number of strain fatigue cycles	Thermal fatigue induced by thermal striping				TS
Fretting wear at contact points on fuel system components	Flow-induced vibration				
Fretting wear at contact points on fuel system components	Hold-down force				
Dimensional changes	Erosion and corrosion buildup				

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Dimensional changes	Pin bundle to duct interaction				TS
Dimensional changes	FA bow/distortion				

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Dimensional changes	FA dilation				TS
Dimensional changes	FA axial growth				

Table 6-4: PIRT for Core Seismic Criteria under Operating Basis Earthquake – Remain operational during and following OBE

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Reactivity Insertion Limit – Unwarranted OBE Scram	Downward force during OBE				TS
Reactivity Insertion Limit – Unwarranted OBE Scram	Lateral displacements during OBE				

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Reactivity Insertion Limit - Post-OBE Operability	Fuel assembly and component residual horizontal deformations				TS
Reactivity Insertion Limit – Post-OBE Operability	Core assembly residual axial displacements				

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Coolant Flow Rate Limit – Post-OBE Operability	Fuel assembly lift-off				TS
Coolant Flow Rate Limit – Post-OBE Operability	Fuel assembly and component residual horizontal displacements				

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Refueling Force Limit – Post-OBE Operability	Fuel assembly and component residual horizontal displacements				TS
Structural Damage Limit – Post-OBE Operability	Fatigue damages				

Table 6-5: PIRT for Core Seismic Criteria under Design Basis Earthquake – No Super-prompt Criticality and Maintain Coolability

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Reactivity Insertion Limit – Pre-scram displacements	Fuel assembly and component horizontal displacements	TS			

Category	Phenomena	Key Influencing Parameters	Importance Ranking	Knowledge Level	Additional Comments
Coolant Flow Rate Limit – Core Coolability	Fuel assembly and component residual deformations				TS

7 FUEL ASSEMBLY QUALIFICATION TASKS

Based on the above sections, the minimum required tasks in the fuel assembly qualification plan is as follows.

1) Core Mechanical Analysis Computer Code Qualification Plan
2) Fuel Assembly Mechanical Test Plan
3) Historical Operational or Pre-existing Experimental Data Qualification Plan
4) Fuel Assembly Design Criteria Evaluation Plan
5) Fuel Assembly Characterization and Surveillance Program

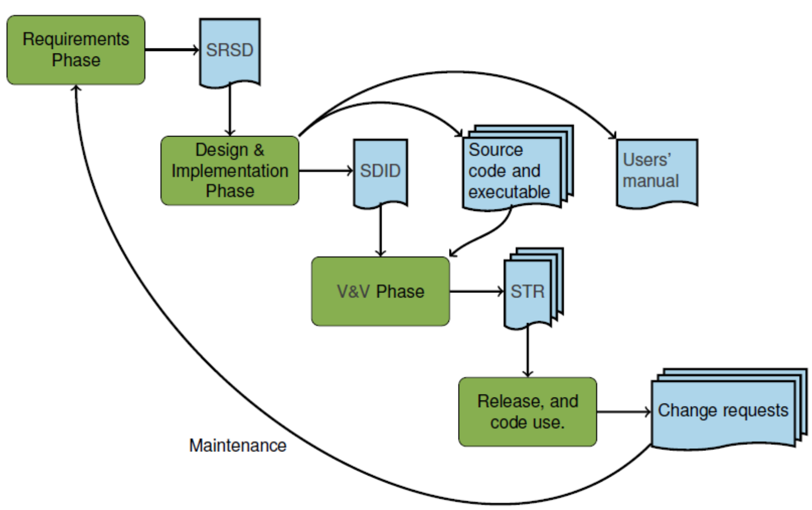
The above tasks are performed as part of a reactor development program. However, the fuel assembly qualification can be completed as an independent program if there's a test reactor or commercial reactor that accommodates full-sized Lead Test Assemblies (LTA) for its operating cycle.

7.1 Core Mechanical Analysis Computer Code Qualification Plan

Computer codes used for the core mechanical analysis can be either developed or acquired from a vendor. The following sections describe the process flow to ensure the computer codes used for the analysis adhere to Subpart 2.7 of ASME NQA-1, "Quality Assurance Requirements for Computer Software for Nuclear Facility Application" and NRC Regulatory Guide 1.168, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants."

7.1.1 Software Development and Verification/Validation

A computer code for core mechanical analysis shall be developed in accordance with a software management plan that complies with the ASME NQA-1. The general software development and maintenance process shall be used, as shown in Figure 7-1.



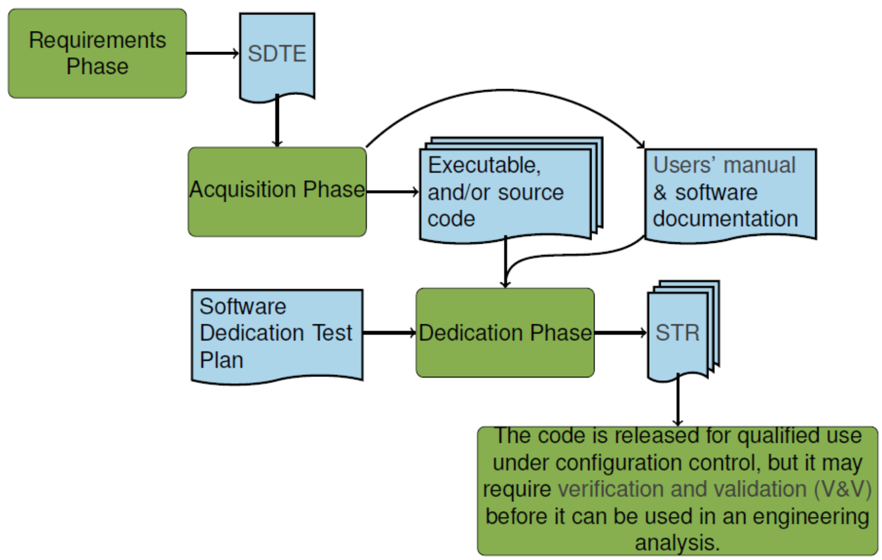
SRSD: Software Requirement and Specification Document
SDID: Software Design Implementation Document
STR: Software Test Report
V&V: Verification & Validation

Figure 7-1: Software Development and Maintenance Process

It is noted that the Software Test Report (STR) includes the Benchmark Comparison Report (BCR). The BCR is used to validate numerical models or analysis results by comparing them with relevant test results (see Section 7.2).

7.1.2 Software Acquisition

Software can also be acquired from a vendor or other sources, as shown in Figure 7-2. If the acquired software was not produced from an ASME NQA-1 qualified vendor, a commercial dedication must be performed as required by the ASME NQA-1, Part II, Subpart 2.14 [11]. It is also noted that software that is acquired and then sufficiently modified is considered developed software.



SDTE: Software Dedication Technical Evaluation

Figure 7-2: Software Acquisition Process

7.2 Fuel Assembly Mechanical Test Plans

Existing experimental data can be used if adequately justified. If no data exists or the existing data is insufficient, a test program should be developed to validate the numerical models of the fuel assembly, which can be categorized into the following areas.

7.2.1 Component Test

Typically, a component-level test is performed to evaluate the structural integrity of a component such as dynamic crush strength, buckling strength, tensile strength, or joint/weld strength. In addition, the following will characterize the mechanical behavior of sub-components.

- Range of motion test of inlet nozzle
- Single fuel pin bend and wrap wire compression test
- Fuel pin bundle bending stiffness
- Pin bundle-to-duct compression test

- Single pin and/or pin bundle flow-induced vibration test
- Hydraulic damping test

7.2.2 Single Fuel Assembly Test

A full-size fuel assembly test shall be performed to characterize the following behaviors. It is noted that a simulated fuel pin bundle or bundle without pins can be used in the test assembly depending on the purpose of test.

- Static load-deflection (mechanical and thermal)
- Natural frequencies and mode shapes
- Lateral and vertical dynamic impact stiffness
- Withdrawal and insertion force

7.2.3 Multiple Fuel Assembly Test

The multiple fuel assembly test is important to support code V&V relating to the core restraint system, fuel handling, and seismic performance. An appropriate hex core configuration, such as 7, 19, or 37 fuel assemblies, will be tested to simulate nearest neighbor interactions of a fuel assembly, and the following mechanical behaviors can be characterized:

- Static load-deflection (mechanical and thermal)
- Withdrawal and insertion force
- Single row core seismic
- Cluster core seismic (e.g., 7assemblies)

7.2.4 Major Effects on Single and Multiple Fuel Assembly Test

It is important to include major effects on fuel assembly behavior such as thermal gradient effects (TE), irradiation effects (IE), and fixity effects (FE) at the boundary condition, as listed in Table 7-1.

Table 7-1: Major Effects on Fuel Assembly Behavior

Type	Descriptions	Test Configurations
Thermal Gradient Effects (TE)	Thermal gradients across a fuel assembly in the lateral direction induces fuel assembly bow	Electric heaters will be attached to the duct outer surfaces
Irradiation Effects (IE)	Fluence gradients across fuel assembly in lateral or vertical directions induce fuel assembly bow and/or dilation	Fuel assembly duct tubes will be pre-deformed as needed
Fixity Effects (FE)	Gap conditions at the boundary condition (i.e., inlet nozzle to receptacle interface) affect fuel assembly rotational stiffness	Normal, loose, and tight gap condition will be used for comparison of projected BOL and EOL conditions

TS

Figure 7-3: Example of Fuel and Control Assembly Mechanical Test Matrix

7.3 Historical Operational or Pre-existing Experimental Data Qualification Plan

A qualification plan may be developed to evaluate and qualify the historical or pre-existing data by using a clear articulation of the following criteria:

- A minimum number of experimental data points
- Calculated uncertainties based on the available data (with a testing plan that collects more data and results in smaller uncertainties)
- Engineering judgment (by individual or panel)
- PIRT results

7.4 Fuel Assembly Design Criteria Evaluation Plan

A comprehensive plan shall be established to provide the fuel assembly design evaluation results that demonstrate all the design basis criteria described in Section 4 are satisfied.

This plan should provide the acceptance criteria, bases, and evaluation methods such as analysis, testing, or comparison for each design requirement. Therefore, this plan should be in harmony with the computer code development and the mechanical test plans. The PIRT will provide a good guideline for prioritizing these activities.

7.5 Fuel Assembly Characterization and Surveillance Program

Fuel surveillance programs shall be considered to add confirmatory data and may be used to address uncertainties associated with fuel characterization data.

7.5.1

TS

7.5.2

TS

TS

7.5.3 Ex-core Surveillance Program

Ex-core surveillance shall be performed on specific core assemblies to obtain detailed behavior and irradiated material properties as a function of loading histories such as burnup, power, location, or temperature until the core assembly's end-of-life (EOL). Due to high assembly dose rates, the surveillance shall occur in a hot cell facility. Specialized inspection/measurement systems, as well as devices to dismantle assemblies, must be utilized. In addition, shipping/handling equipment for spent fuel assemblies must be developed.

Coupons of surveillance materials can be irradiated to the expected lifetime damage dose that critical structural components will experience [13]. Surveillance coupons should be located in test assemblies at locations with higher flux, but in spectra that has nearly the same mean neutron energy as the component of interest, as shown in Figure 7-4 [13]. In addition, instrumentation to obtain the maximum temperature and neutron flux intensity, energy distribution, or spectrum can be placed in the test assemblies.

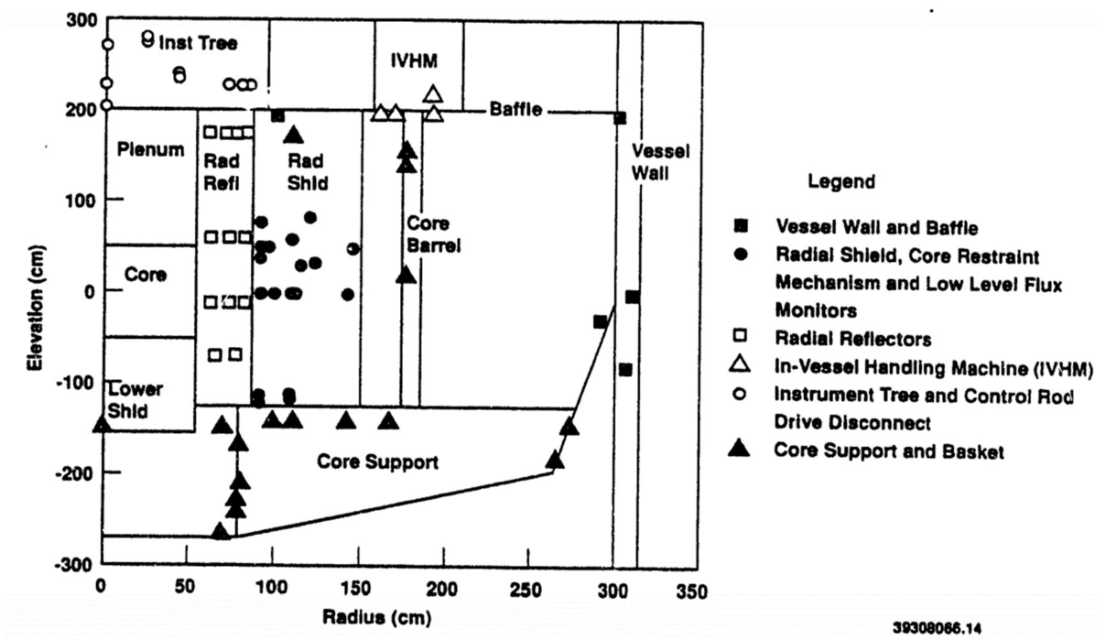


Figure 7-4: Axial and Radial Locations of Critical Structural Components in FFTF

8 REFERENCES

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[4] International Atomic Energy Agency, International Working Group on Fast Reactors, "Verification and Validation of LMFBR Static Core Mechanics Codes - Part I," IWGFR/75, Vienna, Austria.

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[6] International Atomic Energy Agency, "Intercomparison of Liquid Metal Fast Reactor Seismic Analysis Codes Volume 1: Validation of Seismic Analysis Codes Using Reactor Core Experiments," IAEA-TECDOC-798.

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[12] General Electric, "LMFBR In-core Instrumentation and In-vessel Surveillance Systems," U.S. Atomic Energy Commission, GEAP-13825-2, 1972.

[13] Simons, R.L., presented at 8th ASTM-Euratom Symposium on Reactor Dosimetry, "Damage Rates for FFTF Structural Components and Surveillance Assemblies," U.S. Department of Energy, WHC-SA-1746-FP, Vail, CO, USA, Aug 29 - Sept 3, 1993.

END OF DOCUMENT

Enclosure 3

TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))

I, Peter C. Gaillard, hereby state:

1. I am Director, Regulatory Affairs, and I have been authorized by TerraPower, LLC (TerraPower) to review information sought to be withheld from public disclosure in connection with the development, testing, licensing, and deployment of the TerraPower reactor and its associated fuel, structures, systems, and components, and to apply for its withholding from public disclosure on behalf of TerraPower.
2. The information sought to be withheld, in its entirety, is contained in TerraPower's Enclosure 1, which accompanies this Affidavit.
3. I am making this request for withholding and executing this Affidavit as required by 10 CFR § 2.390(b)(1).
4. I have personal knowledge of the criteria and procedures utilized by TerraPower in designating information as a trade secret, privileged, or as confidential commercial or financial information that would be protected from public disclosure under 10 CFR § 2.390(a)(4).
5. TerraPower's information contained in Enclosure 1 accompanying this Affidavit contains non-public details of the TerraPower regulatory and developmental strategies intended to support NRC staff review.
6. Pursuant to 10 CFR § 2.390(b)(4), the following is furnished for consideration by the Commission in determining whether the information in Enclosure 1 should be withheld:
    a. The information has been held in confidence by TerraPower.
    b. The information is of a type customarily held in confidence by TerraPower and not customarily disclosed to the public. TerraPower has a rational basis for determining the types of information that it customarily holds in confidence and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application and substance of that system constitute TerraPower policy and provide the rational basis required.
    c. The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR § 2.390, it is received in confidence by the Commission.
    d. This information is not available in public sources.

e. TerraPower asserts that public disclosure of this non-public information is likely to cause substantial harm to the competitive position of TerraPower, because it would enhance the ability of competitors to provide similar products and services by reducing their expenditure of resources using similar project methods, equipment, testing approach, contractors, or licensing approaches.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 11, 2020

Peter C. Gaillard

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Peter C. Gaillard
Director, Regulatory Affairs
TerraPower, LLC