January 25, 2023

TP-LIC-LET-0051
Project Number 99902100

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

Subject: Submittal of TerraPower Fuel and Control Assembly Qualification Topical Report

This letter transmits the Topical Report, NATD-FQL-PLAN-0004 Revision 0, *TerraPower, LLC (TerraPower) Natrium Topical Report: Fuel and Control Assembly Qualification*, to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The report describes TerraPower's process to obtain qualified fuel assemblies and control assemblies for the Natrium™ reactor, a TerraPower and GE-Hitachi technology. The report describes the current state of TerraPower qualification activities including results that have been completed to date as well as ongoing plans to complete additional activities that will be required to fully qualify the fuel and control assemblies.

TerraPower requests the NRC's review and approval that the qualification methodologies, acceptance criteria, manufacturing parameters, evaluation methods and models, use of legacy data, and planned testing, as described in the enclosed topical report, are adequate to qualify fuel and control assemblies for the Natrium reactor. TerraPower requests that a nominal review duration of one year be considered.

The enclosed topical report contains proprietary and export-controlled information (ECI). It is requested that Enclosure 3, which contains proprietary information, be withheld from public disclosure in accordance with 10 CFR 2.390(a)(4). An affidavit certifying the basis for the request to withhold Enclosure 3 from public disclosure is included as Enclosure 1. Enclosure 3 also contains ECI which must be protected from public disclosure per the requirements of 15 CFR 730 and 10 CFR 810. Proprietary and ECI materials have been redacted from the topical report provided in Enclosure 2; redacted information is identified using [[ ]]$^{(a)(4)}$, [[ ]]^{ECI}, or [[ ]]$^{(a)(4), ECI}$.

This letter and the enclosures make no new or revised regulatory commitments.

If you have any questions regarding this submittal, please contact Ryan Sprengel at rsprengel@terrapower.com or (425) 324-2888.

Sincerely,

Ryan Sprengel
Director of Licensing, Natrium
TerraPower, LLC

Enclosure:

1. TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))
2. TerraPower, LLC Topical Report, NATD-FQL-PLAN-0004, *TerraPower, LLC (TerraPower) Natrium Topical Report: Fuel and Control Assembly Qualification* – Non-Proprietary (Public)
3. TerraPower, LLC Topical Report, NATD-FQL-PLAN-0004, *TerraPower, LLC (TerraPower) Natrium Topical Report: Fuel and Control Assembly Qualification* – Proprietary (Non- Public)

cc: Mallecia Sutton, NRC
William Jessup, NRC
Nathan Howard, DOE
Jeff Ciocco, DOE

ENCLOSURE 1

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

Enclosure 1
TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure
(10 CFR 2.390(a)(4))

I, George Wilson, hereby state:

1. I am the Vice President, 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 Natrium™ 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 Enclosure 3, 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. The information contained in Enclosure 3 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 3 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: January 25, 2023

George Wilson Jr

George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC

ENCLOSURE 2

TerraPower, LLC
Topical Report, NATD-FQL-PLAN-0004
TerraPower, LLC (TerraPower) Natrium Topical Report:
Fuel and Control Assembly Qualification
Non-Proprietary (Public)

NATRIUM
a TerraPower & GE-Hitachi Technology

Document Title TerraPower, LLC (TerraPower) Natrium Topical Report: Fuel and Control Assembly Qualification				
Natrium Document No.: NATD-FQL-PLAN-0004	**Rev. No.:** 0	**Page:** 1 of 119	**Effective Date:** 01/25/2023	**Natrium Safety Class:** SR
Supplier Document No.: N/A	**Supplier Rev:** N/A	**Originating Organization:** NFEQ		**Quality Level:** QL3
Export Controlled: Yes	**Attachments:** N/A			**Document Type:** PLAN
Open Items: N/A	**Record Status:** Released			**Natrium MSL #:** FQL
Approval				
Approval signatures are captured and maintained electronically; See Electronic Approval Records in EDMS.				

REVISION HISTORY

Revision No.	Effective Date	Affected Section(s)	Description of Change(s)
0	01/25/2023	All	Initial Release

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TERMS / ACRONYMS / DEFINITIONS

Acronym	Term/Definition
316 SS	316 Stainless Steel
ABAQUS API	ABAQUS Application Programming Interface. ABAQUS is a commercial finite element analysis code with a scripting interface.
ACCI	Absorber-Cladding Chemical Interaction. Chemical reaction between the cladding and absorber that degrades the cladding mechanical properties. The thickness of the impacted region contributes to cladding wastage.
ACLP	Above Core Load Pad
AFCI	Advanced Fuel Cycle Initiative. Department of Energy program on metallic fuel.
AFQM	Advanced Fuel Qualification Methodology. TerraPower project, funded under a regulatory assistance grant, focused on developing a methodology for qualifying metallic fuel, including early engagement with the NRC.
AOO	Anticipated Operational Occurrences
ASME BPVC	American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Standards for the safe design, manufacture and maintenance of boiler and pressure vessels, power-producing machines, and nuclear power plant components
BCC	Body-centered Cubic. Materials crystal structure
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CRS	Core Restraint System
CSS	Core Support Structure
CTE	Coefficient of Thermal Expansion
DBTT	Ductile to Brittle Transition Temperature
DC	Design Criteria
DE	destructive exams
DOE	Department of Energy
DSC	Differential Scanning Calorimetry
EBR-II	Experimental Breeder Reactor-II. Sodium-cooled fast reactor known for a series of experiments demonstrating passive safety features such as natural convection cooling after a simulated cooling pump failure
EM	Evaluation Model
FCCI	Fuel-Cladding Chemical Interaction. Chemical reaction between the fuel and cladding that degrades the cladding mechanical properties in the interacted zone. The thickness of the impacted region contributes to cladding wastage.
FCRD	Fuel Cycle Research and Development. DOE research program on advanced fuels. They issued a Materials Handbook with relevant materials properties data for the Natrium fuel design.
FEA	Finite Element Analysis
FEM	Finite Element Model
FFTF	Fast Flux Test Facility. 400 MW thermal, liquid sodium cooled fast test reactor that operated from 1982 to 1992
FGR	Fission Gas Release
FIV	Flow Induced Vibration
FM	Ferritic Martensitic

Acronym	Term/Definition
FQAF	Fuel Qualification Assessment Framework
Fuel	Fissile material used to sustain a nuclear chain reaction in a reactor
Fuel Pin	Structural component of sodium-cooled fast reactor that consists of a steel tube housing a fuel column with extra volume (plenum) to contain fission gases
GEH	GE-Hitachi
IAEA	International Atomic Energy Agency
IFR	Integral Fast Reactor. DOE research program to support advancing metallic fuels for closed fuel cycle applications.
IVHM	In-Vessel Handling Machines
LBE	Licensing Basis Events
LDA	Lead Demonstration Assembly. Fuel assemblies with the ability to readily remove fuel pins from the assemblies after irradiation. Used in the Natrium Reactor as part of the fuel surveillance program to provide data on Type 1 fuel at high burnup ahead of the rest of the core, as well as to irradiate lead Type 1B test pins.
LHGR	Linear Heat Generation Rate. Local power generated per unit of length of fuel/absorber.
LMFBR	Liquid Metal Fast Breeder Reactor. Reactor that is cooled by a liquid metal and produces more fissionable material than it consumes to generate energy.
LTA	Lead Test Assembly. Fuel assemblies that contain design features or materials that have not been approved for unrestricted use.
MFF	A series of fuel assemblies with metallic fuel that were irradiated in the FFTF to support conversion of the reactor from mixed oxide fuel to metallic fuel.
NDE	non-destructive exams
NRC	U.S. Nuclear Regulatory Commission
NSMH	Nuclear Systems Materials Handbook
PICT	Peak Inner Cladding Temperature
PIRT	Phenomena Identification and Ranking Tables
PRISM	Power Reactor Innovative Small Module. PRISM is a pool-type, metal-fueled, small modular sodium fast reactor designed by GE-Hitachi
PSAR	Preliminary Safety Analysis Report
RAC	Regulatory Acceptance Criteria. Acceptance criteria derived from regulatory requirements and guidance
RCP	Regulatory Compliance Plan
RES	Reactor Enclosure System
RG	Regulatory Guide
SAS	SAS4A/SASSYS-1 system analysis code
SD	Smear Density. Cross-sectional area of the fresh metallic fuel/cross-sectional area of the fuel pin cladding inner diameter
SFR	Sodium Fast Reactor/Sodium-cooled Fast Reactor. Nuclear reactor with a fast neutron spectrum and liquid sodium coolant
SQA	Software Quality Assurance
SSC	Structure, System, and Component
TLP	Top Load Pad

Acronym	Term/Definition
TREAT	Transient Reactor Test Facility. Test reactor facility at Idaho National Laboratory that can perform extreme transient tests on fuel to assess fuel failure limits and post-failure behavior.
TWR	Traveling Wave Reactor. TerraPower reactor design for a sodium-cooled fast reactor that can convert fertile material into usable fuel through nuclear transmutation, in tandem with the burnup of fissile material. TWRs differ from other kinds of fast-neutron and breeder reactors in their ability to breed and then burn the generated plutonium within the same intact fuel pin, without an interim reprocessing step.
Type 1/ Type 1 Fuel	Fuel utilizing U-10Zr as the fuel alloy, sodium-bond within the fuel, and HT9 cladding. Fuel is similar in composition and dimensions to the fuel pins already reliably used. Natrium Reactor will begin operation with Type 1 fuel.
Type 1B/ Type 1B Fuel	Advanced Natrium Reactor fuel that enables significantly higher burnup
ULOF	Unprotected Loss of Flow
UTOP	Unprotected Transient Over Power
UTS	Ultimate Tensile Strength
V&V	Verification and Validation
YS	Yield Strength

EXECUTIVE SUMMARY

This report presents TerraPower, LLC's (TerraPower) plan to qualify fuel and control assemblies to support operation of the Natrium™ Reactor, a TerraPower and GE-Hitachi technology. A systematic assessment was performed to identify the activities required to support fuel system qualification, including the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of experimental data used to develop and validate evaluation models and empirical safety criteria. This report identifies the acceptance criteria for fuel qualification and presents TerraPower's fuel qualification results to date as well as plans for future fuel qualification activities.

This report includes Regulatory Acceptance Criteria (RAC) (i.e., acceptance criteria derived from regulatory requirements) that when satisfied, support a finding that the fuel is qualified for use (i.e., reasonable assurance exists that the fuel, fabricated in accordance with its specification, will perform as described in the safety analysis). Specifically, the fuel design criteria and associated limits ensure four key objectives: 1) the fuel system is not damaged as a result of normal operation and Anticipated Operational Occurrences (AOOs), 2) the number of fuel pin failures is not underestimated for postulated accidents, 3) coolability is always maintained, and 4) fuel system damage is never so severe during postulated accidents as to prevent reactivity control and control rod insertion when it is required. High-importance fuel phenomena identified for all applicable fuel pin design limits include fission gas release, HT9 mechanical behavior as a function of environmental conditions, fuel-cladding chemical interaction, and fuel thermal conductivity as a function of irradiation/porosity.

Completed and ongoing efforts address major aspects of fuel qualification requirements, while future analyses (e.g., fretting and fatigue behavior, additional testing and analysis to address extreme transients) are expected to provide the final scope of information needed to fully qualify fuel for the Natrium Reactor. Fuel qualification for the Natrium Reactor relies, in part, on historic operating experience and historic data (e.g., Experimental Breeder Reactor-II (EBR-II) and Fast Flux Test Facility (FFTF) metallic fuel pins). This historic data will be qualified under a program that satisfies the quality assurance requirements of 10 CFR 50 Appendix B. With no operating fast-spectrum reactor available to perform final tests, a surveillance program is proposed to monitor the irradiation performance of the fuel to ensure consistent performance with historic operating experience and analytical predictions. The proposed surveillance program includes the capabilities to incorporate knowledge gained from analyses and testing data that becomes available as fuel qualification activities progress.

ACKNOWLEDGEMENT

This topical report represents the effort and determination of many people, including the following contributors: Jesse Cheatham, Ryan Christensen, Francesco Deleo, Lynne Ecker, Ian Gifford, Bruce Hilton, Joseph Hoffman, Virginia Hollis, Julie Jordan, Joseph LaPrad, Ahn Mai, Jason Meng, Sam Miller, Brian Morris, Drew Mueller, Matthew Presson, Christopher Regan, and Javier Romero.

1. PURPOSE

This report presents the TerraPower plan to qualify fuel to support operation of the Natrium Reactor. The overall fuel qualification approach (planning, testing, analysis, etc.) used to obtain qualified fuel is described. TerraPower's fuel qualification efforts have been informed by U.S. Nuclear Regulatory Commission (NRC) guidance, including Regulatory Guide (RG) 1.206, Section C.I.4, “Reactor” [1], NUREG-0800, Section 4.2, “Fuel System Design” [2], and NUREG-2246, “Fuel Qualification for Advanced Reactors” [3]. Additionally, principal design criteria (PDC) that are applicable to fuel performance and fuel qualification have been informed by RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors” [4]. TerraPower has provided several reports to the NRC regarding fuel qualification efforts [2] [5] [7], and the NRC has provided feedback [3] [4] [6]; the NRC's feedback has informed the development of TerraPower's overall approach to fuel qualification. The information presented in this report will apply to licensing efforts associated with the Natrium Reactor design.

This report identifies the RAC (i.e., acceptance criteria derived from regulatory requirements) that will be used for fuel qualification and presents TerraPower's fuel qualification results to date. Fuel qualification for the Natrium Reactor design includes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of experimental data used to develop and validate evaluation models and empirical safety criteria. TerraPower uses historic operating experience and data from EBR-II and FFTF, verifying the suitability of the historic data and qualifying the historic data for use in TerraPower's fuel qualification methodology. This report includes RAC that when satisfied, support a finding that the fuel is qualified for use (i.e., reasonable assurance exists that the fuel, fabricated in accordance with its specification, will perform as described in the safety analysis). Specifically, the fuel design criteria and associated limits must ensure four key objectives: 1) the fuel system is not damaged as a result of normal operation and AOOs, 2) the number of fuel pin failures is not underestimated for postulated accidents, 3) coolability is always maintained, and 4) fuel system damage is never so severe during postulated accidents as to prevent reactivity control and control rod insertion when it is required.

The objective of the Natrium Reactor fuel qualification plan is to confirm that all aspects of the fuel system design and fabrication process will provide reliable and safe operation of a commercial sodium-cooled, fast-neutron spectrum nuclear reactor. This document provides information to the NRC to qualify the fuel for the Natrium Reactor. NRC's review and approval are requested for the following:

- The identified acceptance criteria are adequate to support fuel qualification.
- The identified key fuel manufacturing parameters are adequate to support fuel qualification.
- The identified evaluation methods and models are adequate to support fuel qualification.
- The use of legacy data and the planned testing is adequate to provide the necessary information to qualify the fuel.
- The plans for inclusion of small subsets of fuel pins that operate outside the performance envelope of the bulk of the core, or that feature advanced design features, are acceptable.

2. BACKGROUND

2.1 Design Background

TerraPower's Natrium Reactor is a sodium-cooled fast reactor (SFR) that uses a fuel design and an operating environment that are significantly different from light water reactors currently utilized in the United States. The Natrium Reactor is an innovative design that facilitates rapid construction and achieves cost competitiveness and flexible operations through the adoption of new technology and a reimagined plant layout. Many of these advances are enabled through inherent safety features of pool-type SFRs [5]. The Natrium Reactor design is based on early reactor technology developed in the US by the Department of Energy (DOE) and was developed from decades of research, design, and development from GE-Hitachi's (GEH) Power Reactor Innovative Small Module (PRISM) technology and TerraPower's Traveling Wave Reactor (TWR®) technology. The nuclear heat source is a pool-type sodium fast reactor design with sodium-bonded uranium-10wt% zirconium (U-10Zr) fuel clad in HT9 stainless steel. The reactor operates at about atmospheric pressure, circulating sodium through its core, with heat transferred from the primary sodium to an intermediate sodium loop. The Natrium design uses sodium-bonded metallic fuel consistent with the HT9-clad fuel used successfully in both EBR-II and the FFTF (see Figure 2-1).

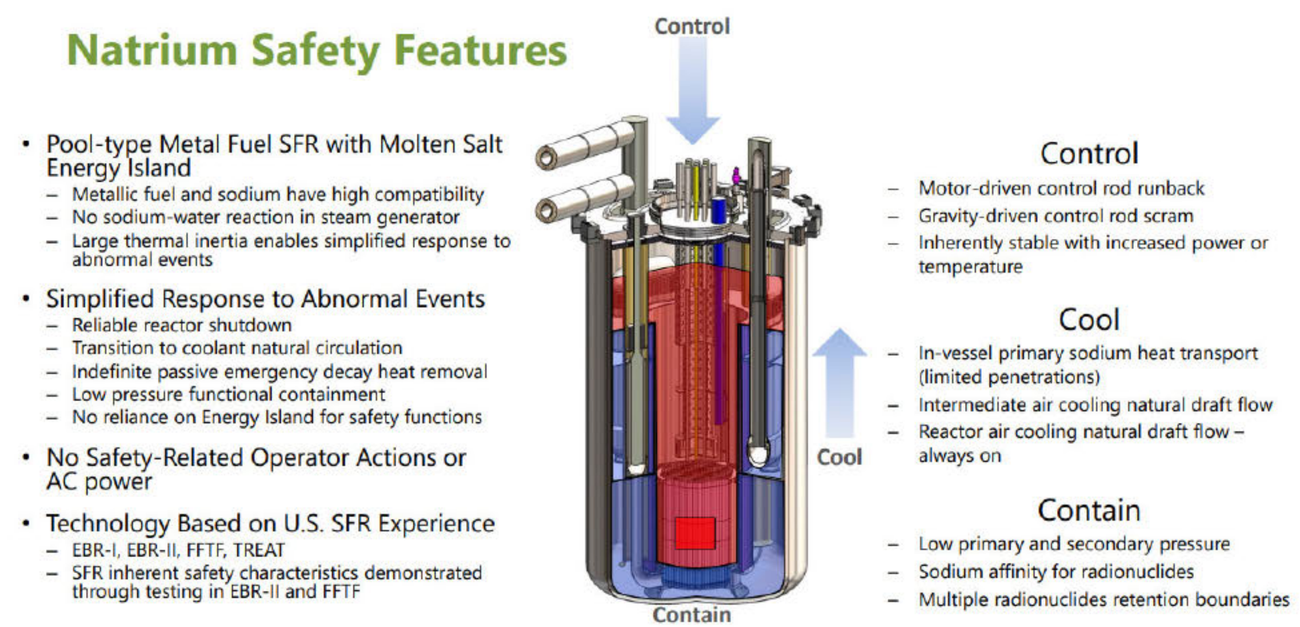

Figure 2-1. Overview of Natrium Reactors' Safety Features

Despite the advanced design, in many respects, the Natrium Reactor takes an incremental approach to design and licensing. The Natrium Reactor will begin operation with Type 1 fuel that is similar in composition and dimensions to the fuel pins reliably used in the EBR-II and the FFTF. The operating conditions such as temperature, mechanical loads and burnup are also within the experience base obtained with previous SFRs. The core has the capability to safely irradiate fuel pins to higher burnup in specific core locations in Lead Demonstration Assemblies (LDAs). The surveillance and sampling of fuel pins in the LDAs will provide data to extend the residence time and burnup of Type 1 fuel pins. The core will also have Lead Test Assemblies (LTAs) to gain operating experience with

Type 1B fuel. Type 1B fuel incorporates advanced features [[ ]]$^{(a)(4)(ECI)}$ that will enable significantly higher fuel burnups and improved fuel utilization to reduce refueling. This document focuses largely on the Type 1 fuel design but will touch on the LDA and LTA designs due to their importance to the fuel surveillance program as well as supporting rapid transition to more advanced fuel designs.

2.2 Regulatory Background

TerraPower submitted the Advanced Fuel Qualification Methodology (AFQM) Report to the NRC on July 16, 2020 (ML20209A155) [6]. The AFQM Report describes methodologies, regulatory criteria, and qualification criteria for metallic fuel for SFRs. The NRC provided feedback in a November 19, 2020 letter, “NRC Feedback Regarding TerraPower White Paper “Advanced Fuel Qualification Methodology Report-Regulatory Guidance Development Report” (EPID No.: L-2020-LRO-0045)” (ML20310A278) [7]. The NRC’s feedback has informed the development of TerraPower’s overall approach to fuel qualification.

TerraPower submitted the Advanced SFR Fuel Assembly Qualification Plan to the NRC by letter dated November 11, 2020 (ML20316A038). The NRC provided feedback in a May 4, 2021 letter, “U.S. Nuclear Regulatory Commission Feedback Regarding TerraPower, LLC’s Advanced Sodium Fast Reactor Fuel Assembly Qualification Plan (EPID NO.: L-2020-LRO-0080)” (ML21099A081) [8]. The NRC’s feedback has informed the development of TerraPower’s fuel qualification efforts for the integrated fuel assembly.

By letter dated February 26, 2021 (ML21057A008), TerraPower submitted the Advanced SFR Type 1 Fuel Pin Qualification Plan to the NRC [9]. The NRC provided feedback in a July 13, 2021 letter, “TerraPower, LLC - U.S. Nuclear Regulatory Commission Staff Feedback Regarding White Paper, “Advanced SFR Type 1 Fuel Pin Qualification Plan”, Revision 0 (EPID NO.: L-2021-LRO-0008)” (ML21147A548) [10]. The NRC’s feedback has informed the development of TerraPower’s fuel qualification efforts for fuel pin specific aspects.

TerraPower’s fuel qualification efforts have been informed by the NRC’s feedback as described above, as well as NRC guidance including RG 1.206, Section C.1.4, “Reactor,” NUREG-0800, Section 4.2, “Fuel System Design,” [2] and NUREG-2246, “Fuel Qualification for Advanced Reactors.” [3]

TerraPower’s fuel qualification efforts began, in part, by identifying RAC that were developed using the guidance of RG 1.206 and NUREG-0800, with adaption as necessary due to the differences from light water reactor technology. Subsequently, the NRC issued NUREG-2246, “Fuel Qualification for Advanced Reactors,” which includes fuel qualification assessment framework (FQAF) goals. Table 2-1 provides a cross-reference between the TerraPower developed/identified RAC and the NUREG-2246 FQAF goals, identifying which RAC are used to address specific FQAF goals. In several cases, FQAF goals are addressed by design specifications as identified in Table 2-1.

Table 2-1. TerraPower Identified/Developed RAC Mapped to NUREG-2246 Appendix A Goals in FQAF

FQAF Goal ID	FQAF Goal Description	RAC #	RAC / Design Specification Description
G1	Fuel is manufactured in accordance with a specification	4.2-5	The fuel system description and design drawings shall provide information necessary to verify that the fuel system design bases are met.
G1.1	Key dimensions and tolerances of fuel components are specified	4.2-5	Relevant key dimensions and tolerances of fuel components are specified in design drawings. Section 5.6 of this report summarizes applicable drawings/drawing types that will be the primary sources for specifying the key dimensions and tolerances identified by RAC 4.2-5.
G1.2	Key constituents are specified with allowance for impurities	4.2-5	Relevant key constituents with allowance for impurities are specified in fuel, material, and product specifications. Section 5.6 of this report summarizes applicable specifications that will be the primary sources for specifying the key constituent and impurity limits identified by RAC 4.2-5.
G1.3	End state attributes for materials within fuel components are specified or otherwise justified	4.2-5	End state attributes (i.e., microstructure, heat treatments, and specific manufacturing processes) are specified in fuel, material, and product specifications. Section 5.6 of this report summarizes applicable specifications that will be the primary sources for specifying the key end state attributes identified by RAC 4.2-5.
G2	Margin to safety limits can be demonstrated	4.2-1, 4.2-2, 4.2-3, 4.2-4	Design criteria and evaluation methods are described below for the subgoals of G2.
G2.1	Margin to design limits can be demonstrated under conditions of normal operation and AOOs	4.2-1	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.
G2.1.1	Fuel performance envelope is defined	4.2-1.1	Stress, strain, or loading limits for all fuel system components shall be established.
		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.
		4.2-1.3	Limits on fretting wear at contact points on all fuel system components shall be established or alternatively, impacts of fretting wear shall be explicitly assessed when demonstrating compliance with fuel system damage criteria that may be affected by fretting wear.
		4.2-1.4	Limits on erosion and corrosion shall be established for all fuel system components or

FQAF Goal ID	FQAF Goal Description	RAC #	RAC / Design Specification Description
			alternatively, impacts of erosion and corrosion shall be explicitly assessed when demonstrating compliance with fuel system damage criteria that may be affected by erosion and corrosion.
		4.2-1.5	Limits on internal cladding damage (wastage) due to fuel-cladding chemical interaction (FCCI) with fuel or absorber-cladding chemical interaction (ACCI) for absorber components shall be established or alternatively, impacts of wastage shall be explicitly assessed when demonstrating compliance with fuel system damage criteria that may be affected by wastage.
		4.2-1.6	Limits on fuel dimensional changes, such as fuel 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.
		4.2-1.7	Limits on dimensional changes, such as absorber pin bowing, control 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.
		4.2-1.8	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 that may be affected by pin internal pressure.
		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 assemblies.
		4.2-1.10	The worst-case hydraulic loads for normal operation and AOOs shall not exceed the hold-down capability of a reactivity control assembly.
		4.2-1.11	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.
		4.2-1.12	Design temperature limits on fuel system components for normal operation and AOOs

FQAF Goal ID	FQAF Goal Description	RAC #	RAC / Design Specification Description
			shall be established, or alternatively, peak temperature shall be explicitly assessed in analyses demonstrating compliance with fuel system damage criteria that may be affected by temperature.
G2.1.2	Evaluation model is available (see EM Assessment Framework)	4.2-6	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. Section 6 provides more details on the approach to design evaluations with specific discussion of evaluation models in Section 6.3.2.2.
G2.2	Margin to radionuclide release limits under accident conditions can be demonstrated	4.2-2	Fuel pin failure criteria shall be established that ensure that the number of fuel pin failures cannot be underestimated for all failure mechanisms that may result in the loss of fuel integrity (cladding breach) during normal operation, AOOs, and postulated accidents.
G2.2.1	Radionuclide retention requirements are specified		Radionuclide retention requirements will be described in Chapter 2, “Methodologies and Analysis”, of the Natrium Preliminary Safety Analysis Report (PSAR). Fuel failure criteria and fuel performance methods, which are used to demonstrate that radionuclide retention requirements are met, will also be described in Chapter 2 of the PSAR.
G2.2.2	Criteria for barrier degradation and failure are suitably conservative (a) Criteria are conservative	4.2-2.1	Fuel system design limits shall be established and used for the prediction of fuel pin failure due to overheating of the cladding or alternatively, fuel pin failure due to overheating of the cladding shall be explicitly assessed in analyses demonstrating compliance with fuel failure criteria that may be affected by fuel pin overheating of the cladding.
		4.2-2.2	Fuel system design limits shall be established and used for the prediction of fuel pin failure due to overheating of the fuel slug or alternatively, fuel pin failure due to overheating of the fuel slug shall be explicitly assessed in analyses demonstrating compliance with fuel failure criteria that may be affected by overheating of the fuel slug.
		4.2-2.3	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 or alternatively, deformation of the cladding from mechanical load shall be explicitly assessed in

FQAF Goal ID	FQAF Goal Description	RAC #	RAC / Design Specification Description
			analyses demonstrating compliance with fuel failure criteria that may be affected by deformation of the cladding.
		4.2-2.4	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.
		4.2-2.5	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 or alternatively, cladding wastage shall be explicitly assessed in analyses demonstrating compliance with fuel failure criteria that may be affected by cladding wastage.
	(b) Experimental data are appropriate (see ED Assessment Framework)	4.2-6	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. Section 6 provides more details on the approach to design evaluations with specific discussion of test data in 6.2 and 6.3.
G2.2.3	Radionuclide retention and release from fuel matrix are modeled conservatively (a) Model is conservative (b) Experimental data are appropriate (see ED Assessment Framework)	4.2-6	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. Section 6 provides more details on the approach to design evaluations with specific discussion of test data in 6.2 and 6.3.
G2.3	Ability to achieve and maintain safe shutdown is assured	4.2-4	Reactivity control assembly criteria shall be established for all damage mechanisms that may occur during postulated accidents to ensure that control rods can be fully inserted when required.
G2.3.1	Coolable geometry is ensured	4.2-3	Fuel assembly criteria shall be established for all 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.
	(a) Criteria to ensure coolable geometry are specified	4.2-3.1	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 or alternatively, cladding stress and strain during postulated accidents shall be explicitly assessed in analyses demonstrating

FQAF Goal ID	FQAF Goal Description	RAC #	RAC / Design Specification Description
			compliance with fuel coolability criteria that may be affected by cladding stress and strain during postulated accidents.
		4.2-3.2	The maximum temperature of the cladding during postulated accidents shall be less than the melting temperature of the cladding.
		4.2-3.3	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 (i.e., swelling) of the cladding during postulated accidents.
		4.2-3.4	The maximum temperature of the fuel slug during postulated accidents shall be less than the melting temperature of the fuel.
		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.
		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.
	(b) Evaluation models are available (see EM Assessment Framework	4.2-6	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. Section 6 provides more details on the approach to design evaluations with specific discussion of evaluation models in Section 6.3.2.2.
G2.3.2	Negative reactivity insertion can be demonstrated	4.2-4	Reactivity control assembly criteria shall be established for all damage mechanisms that may occur during postulated accidents to ensure that control rods can be fully inserted when required.
	(a) Criteria are provided to ensure that negative reactivity insertion is not obstructed	4.2-4.1	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.
		4.2-4.2	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.

FQAF Goal ID	FQAF Goal Description	RAC #	RAC / Design Specification Description
	(b) Evaluation model is available (see EM Assessment Framework)	4.2-6	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. Section 6 provides more details on the approach to design evaluations with specific discussion of evaluation models in Section 6.4

As can be seen in Table 2-1, FQAF goals are addressed directly by RAC with the exception of FQAF goal G2.2.1 Radionuclide Retention Requirements are Specified. Specific radionuclide retention requirements will be specified in Chapter 2, "Methodologies and Analysis," of the Natrium PSAR, with the fuel failure criteria and fuel performance methods demonstrating that the radionuclide retention requirements are met. In addition to the overall FQAF goals, NUREG-2246 identifies Assessment Framework goals for evaluation models as well as for supporting experimental data. These goals and their correspondence to associated RAC/Design Specifications are summarized in Table 2-2 and Table 2-3, respectively. All of the evaluation model and experimental data assessment framework goals are addressed by RAC 4.2-6. RAC 4.2-6 specifies that: "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". Design evaluations are further clarified to include operating experience, testing, and analytical predictions. The "Compliance Specific Considerations" of RAC 4.2-6 provide significantly more detail on expectations for acceptable design evaluation methods, but a high-level prescribed expectation is that they apply conservative treatment of uncertainties in the values of important parameters. Because RAC 4.2-6 does not explicitly address all of the Evaluation Model and Experimental Data Assessment Framework goals, more details are provided relative to plans to address the goals in Section 6, where the plans for fuel system design evaluation are discussed.

Table 2-2. TerraPower Identified/Developed RAC Mapped to NUREG-2246 Evaluation Model Assessment Framework Goals

FQAF Goal ID	Evaluation Model Assessment Framework Goal Description	RAC #	RAC / Design Specification Description
EM G1	Evaluation model contains the appropriate modeling capabilities	4.2-6	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. Section 6 provides more details on the approach to design evaluations with specific discussion of evaluation models in Section 6.3.2.2.
EM G1.1	Evaluation model is capable of modeling the geometry of the fuel system		
EM G1.2	Evaluation model is capable of modeling the material properties of the fuel system		
EM G1.3	Evaluation model is capable of modeling the physics relevant to fuel performance		
EM G2	Evaluation model has been adequately assessed against experimental data		

FQAF Goal ID	Evaluation Model Assessment Framework Goal Description	RAC #	RAC / Design Specification Description
EM G2.1	Data used for assessment are appropriate (see ED Assessment Framework)		
EM G2.2	Evaluation model is demonstrably able to predict fuel failure and degradation mechanisms over the test envelope		
EM G2.2.1	Evaluation model error is quantified through assessment against experimental data		
EM G2.2.2	Evaluation model error is determined throughout the fuel performance envelope		
EM G2.2.3	Sparse data regions are justified		
EM G2.2.4	Evaluation model is restricted to use within its test envelope		

Table 2-3. TerraPower Identified/Developed RAC Mapped to NUREG-2246 Experimental Data Assessment Framework Goals

FQAF Goal ID	Experimental Data Assessment Framework Goal Description	RAC #	RAC / Design Specification Description
ED G1	Assessment data are independent of data used to develop/train the evaluation model	4.2-6	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. Section 6 provides more details on the approach to design evaluations with specific discussion of evaluation models in Section 6.3.2.2.
ED G2	Data has been collected over a test envelope that covers the fuel performance envelope		
ED G3	Experimental data have been accurately measured		
ED G3.1	The test facility has an appropriate quality assurance program		
ED G3.2	Experimental data are collected using established measurement techniques		
ED G3.3	Experimental data account for sources of experimental uncertainty		
ED G4	Test specimens are representative of the fuel design		
ED G4.1	Test specimens are fabricated consistent with the fuel manufacturing specification		
ED G4.2	Distortions are justified and accounted for in the experimental data		

3. DISCUSSION

Pool-type SFRs were constructed as early as 1951 (e.g., Experimental Breeder Reactor–I in the U.S.A.) and the fuel system and fuel pin designs evolved based on operating experience and transient testing until the early 1990s. During the fast reactor development programs, fuel pins fabricated with various fuel-cladding material combinations and over a range of dimensions were shown to have excellent reliability. Well over 100,000 metallic fuel pins were used to run fast reactors with an exceptionally low failure rate. Excellent transient behavior was also demonstrated for fuel pins both using the EBR-II to run full core transients that represent accident scenarios and the Transient Reactor Test (TREAT) facility to run overpower transients up to (and even exceeding) four times the nominal power (i.e., ≥400% nominal power).

A Regulatory Compliance Plan (RCP) has been developed that identifies and adapts the regulatory requirements that are described in Section 4.2 of the Standard Review Plan (NUREG-0800), which is devoted to the fuel system design [11]. This adaptation did not only update the terminology to be more suited for metallic fuel in SFRs but also included specific phenomena of concern for this fuel system based on extensive review of the available data and historic operating experience. The referenced RCP [11] also establishes RAC to ensure compliance with the identified regulatory requirements. Fuel and absorber design criteria and associated limits and bases were provided in the Natrium Fuel, Control, Shield, and Reflector Pin Design Basis [12] to ensure compliance of fuel and absorber pin designs with the established RAC. A key fuel testing need is to demonstrate the suitability of the established fuel design criteria and limits to prevent damage and/or failure, maintain coolability of the core during and after all licensing basis events (LBEs), and assure fuel system damage during postulated accidents will not prevent reactivity control rod insertion when required. Extensive review of public and non-public data has been performed when establishing these fuel design criteria and limits. References to some of the compilations of data used to establish these criteria and limits are included in Section 6, with additional proposed activities to address any gaps summarized in Section 6.3. Beyond establishing the proper fuel design criteria, demonstrating compliance of fuel with these design criteria for the applicable operating domains is another important task to support licensing. Due to the inherently complex nature of nuclear fuels, multiple physical phenomena must be adequately modeled to provide reliable predictions of fuel pin behavior. Test data are required over applicable ranges for high-importance phenomena to validate sufficient understanding of these phenomena and overall reliability of the associated fuel models.

This assessment is organized to be roughly consistent with the RCP [11], capturing applicable information and planned activities to address the key areas of review for nuclear fuel system designs (see Figure 3-1 for a flow chart). Specifically, 1) Fuel Design Criteria, 2) Fuel System Description, 3) Design Evaluation, 4) Testing and Inspection of New Fuel, 5) Online Fuel System Failure Monitoring, and 6) Post Irradiation Surveillance Plans will be addressed. The primary emphasis will be on the Fuel Design Criteria and Design Evaluation aspects of the RCP since they are the most dependent on testing support prior to the startup of an advanced reactor. The more detailed test plans developed in subsequent efforts will evaluate the applicable operating range of the Natrium Reactor, available applicable data, additional data needed to cover the most adverse conditions anticipated, and the number of data points required to reduce associated uncertainties to acceptable levels.

Fuel Regulatory Acceptance Criteria	• Fuel Regulatory Acceptance Criteria (RAC) from existing NRC regulations were identified and adapted to metallic fuel in Regulatory Compliance Plan • Largely adapted from NUREG-0800 content to apply to specific concerns for metallic fuel
Fuel Design Criteria	• Verify corresponding fuel design criteria have been established to address each RAC • Review and cite applicable data to justify suitability of criteria and associated limits
Fuel System Description	• Reviews and updates typical content expected to define the fuel system design to apply to metallic fuel • Cite approach to document/address each of the identified aspects of the fuel design
Design Evaluation	• Phenomena Identification Ranking Tables (PIRT) developed for each of the fuel design criteria to identify high importance phenomena and knowledge level to prioritize efforts • For each fuel design criteria identify applicable methods to demonstrate compliance with design limits (Operating Experience, Testing, Analytical Predictions)
Testing & Inspection of New Fuel	• Document identified testing and inspections expected for new fuel
Online Fuel System Failure Monitoring	• Identify equipment, design features, and approach to detect fuel failures during operation
Post Irradiation Surveillance	• Based on the PIRT assessment and planned testing and analysis activities a surveillance program within the Natrium™ reactor is planned to address remaining uncertainties

Figure 3-1. Overall Fuel Qualification Assessment Logic Flow

4. FUEL DESIGN CRITERIA

Fuel design criteria must be set to achieve four key objectives: 1) the fuel system is not damaged as a result of normal operation and AOOs, 2) the number of fuel pin failures is not underestimated for postulated accidents, 3) coolability is always maintained, and 4) fuel system damage is never so severe during postulated accidents as to prevent reactivity control rod insertion when it is required. As stated above, a key testing and analysis need is to demonstrate the suitability of the established fuel design criteria and associated limits. To help ensure adequate coverage of each of the established fuel design criteria, Table 4-1 through Table 4-4 summarize the existing RAC and associated design criteria. Table 4-1 covers design basis criteria to prevent fuel damage; Table 4-2 addresses criteria to predict fuel failure; Table 4-3 outlines criteria to maintain fuel coolability; and Table 4-4 addresses criteria to ensure reactivity control insertability. Additional testing activities have been identified to supplement the currently available data to further justify established design basis limits, but these are discussed in Section 6.3, where the applicable Testing Activities are summarized.

Table 4-1. Design Criteria to Prevent Fuel System Damage-

Specific RAC	Acceptance Criterion	Fuel Pin Applicable Design Criteria	Fuel Assembly Applicable Design Criteria	Absorber Pin Applicable Design Criteria	Control Assembly Applicable Design Criteria
4.2-1.1	Stress, strain, or loading limits for all fuel system components shall be established.	[[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4)
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.	[[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4)
4.2-1.3	Limits on fretting wear at contact points on all fuel system components shall be established or alternatively, impacts of fretting wear shall be explicitly assessed when demonstrating compliance with fuel system damage criteria that may be affected by fretting wear.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-1.4	Limits on erosion and corrosion shall be established for all fuel system components or alternatively, impacts of erosion and corrosion shall be explicitly assessed when demonstrating compliance with fuel system damage criteria that may be affected by erosion and corrosion.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)

Specific RAC	Acceptance Criterion	Fuel Pin Applicable Design Criteria	Fuel Assembly Applicable Design Criteria	Absorber Pin Applicable Design Criteria	Control Assembly Applicable Design Criteria
4.2-1.5	Limits on internal cladding damage (wastage) due to fuel-cladding chemical interaction (FCCI) with fuel or absorber-cladding chemical interaction (ACCI) for absorber components shall be established or, alternatively, impacts of wastage shall be explicitly assessed when demonstrating compliance with fuel system damage criteria that may be affected by wastage.	[[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-1.6	Limits on dimensional changes, such as fuel 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.	[[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-1.7	Limits on dimensional changes, such as absorber pin bowing, control 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.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-1.8	Design limits on fuel pin and reactivity control absorber pin internal pressure for normal operation and AOOs shall be	[[	[[ ]](a)(4)	[[	[[ ]](a)(4)

Specific RAC	Acceptance Criterion	Fuel Pin Applicable Design Criteria	Fuel Assembly Applicable Design Criteria	Absorber Pin Applicable Design Criteria	Control Assembly Applicable Design Criteria
	established or alternatively, pin internal pressure shall be explicitly assessed in analyses demonstrating compliance with fuel system damage criteria that may be affected by pin internal pressure.	]](a)(4)		]](a)(4)	
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.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-1.10	The worst-case hydraulic loads for normal operation and AOOs shall not exceed the hold-down capability of a reactivity control assembly.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-1.11	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.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4)
4.2-1.12	Design temperature limits on fuel system components for normal operation and AOOs shall be established, or alternatively, peak temperature shall be explicitly assessed in analyses demonstrating compliance with fuel system damage criteria that may be affected by temperature.	[[ ]](a)(4) [[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4)

Table 4-2. Design Criteria to Prevent Fuel System Failure

Specific RAC	Acceptance Criterion	Fuel Pin Applicable Design Criteria	Fuel Assembly Applicable Design Criteria	Absorber Pin Applicable Design Criteria	Control Assembly Applicable Design Criteria
4.2-2.1	Fuel system design limits shall be established and used for the prediction of fuel pin failure due to overheating of the cladding or alternatively, fuel pin failure due to overheating of the cladding shall be explicitly assessed in analyses demonstrating compliance with fuel failure criteria that may be affected by fuel pin overheating of the cladding.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-2.2	Fuel system design limits shall be established and used for the prediction of fuel pin failure due to overheating of the fuel slug or alternatively, fuel pin failure due to overheating of the fuel slug shall be explicitly assessed in analyses demonstrating compliance with fuel failure criteria that may be affected by overheating of the fuel slug.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-2.3	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 or, alternatively, deformation of the cladding from mechanical load shall be explicitly assessed in analyses demonstrating compliance with fuel failure criteria that may be affected by deformation of the cladding.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-2.4	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.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-2.5	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 or alternatively, cladding wastage shall be explicitly assessed in analyses demonstrating	[[ ]](a)(4) [[	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)

Specific RAC	Acceptance Criterion	Fuel Pin Applicable Design Criteria	Fuel Assembly Applicable Design Criteria	Absorber Pin Applicable Design Criteria	Control Assembly Applicable Design Criteria
	compliance with fuel failure criteria that may be affected by cladding wastage.	]](a)(4)			

Table 4-3. Design Criteria to Ensure Fuel Coolability

Specific RAC	Acceptance Criterion	Fuel Pin Applicable Design Criteria	Fuel Assembly Applicable Design Criteria	Absorber Pin Applicable Design Criteria	Control Assembly Applicable Design Criteria
4.2-3.1	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 or alternatively, cladding stress and strain during postulated accidents shall be explicitly assessed in analyses demonstrating compliance with fuel coolability criteria that may be affected by cladding stress and strain during postulated accidents.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-3.2	The maximum temperature of the cladding during postulated accidents shall be less than the melting temperature of the cladding.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-3.3	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 (i.e., swelling) of the cladding during postulated accidents.	[[ ]](a)(4)	[[ ]](a)(4) [[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-3.4	The maximum temperature of the fuel slug during postulated accidents shall be less than the melting temperature of the fuel.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)

Specific RAC	Acceptance Criterion	Fuel Pin Applicable Design Criteria	Fuel Assembly Applicable Design Criteria	Absorber Pin Applicable Design Criteria	Control Assembly Applicable Design Criteria
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.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
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.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)

Table 4-4. Design Criteria to Ensure Reactivity Control Insertability Criteria

Specific RAC	Acceptance Criterion	Fuel Pin Applicable Design Criteria	Fuel Assembly Applicable Design Criteria	Absorber Pin Applicable Design Criteria	Control Assembly Applicable Design Criteria
4.2-4.1	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.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
4.2-4.2	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.	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)

5. FUEL DESIGN DESCRIPTION

5.1 Overview of the Fuel Design of the Natrium Reactor

The Natrium Reactor is a pool-type, sodium-cooled, fast-spectrum reactor with some design similarities to other SFRs such as EBR-II and the FFTF. The Natrium Reactor core will contain Type 1 fuel at the beginning of life. Natrium Type 1 fuel pins are intentionally similar in design to historically tested designs to leverage available historic operating experience. Specifically, they use U-10Zr, sodium bonding, HT9 cladding, and a nominal smear density of 75%. Type 1 fuel has a cladding diameter similar to fuel pins that were successfully tested in EBR-II ([[

]]$^{(a)(4)}$. Bundles of Type 1 fuel pins are located in hex-shaped fuel assemblies in the core similar to EBR-II and FFTF; however, a larger number of pins will be used per assembly than was for FFTF or EBR-II. A comparison between fuel pin dimensions from various reactors is given in Table 6-5. Regarding operational conditions, the Natrium fuel operates at low power [[
]]$^{(a)(4)}$. More explicit comparison to historic designs and targeted operating conditions are provided in Section 6.2, Historic Operating Experience.

In addition to standard fuel assemblies, the Natrium Reactor core will have Lead Demonstration Assemblies (LDA) that are a crucial component of the fuel surveillance and Lead Test Assembly (LTA) programs. The LDAs are designed to expedite the availability of post-irradiated data on fuel pins by providing removable fuel pins that [[

]]$^{(a)(4)}$. The targeted conditions for the accelerated LDA pins will still be bound by historic pin operating experience and fuel performance assessments to verify design limits are met. More detailed discussion of the planned Fuel Surveillance Program is provided in Section 9. Although the primary purpose of the LDAs is to support Type 1 fuel surveillance, [[

]]$^{(a)(4)}$.

The basic nuclear control component of the Natrium Reactor core is the Control Assembly which contains absorber pins. The absorber pins contain cladded, helium-bonded, boron carbide (B_4C) absorber pellets that can be adjusted in the axial direction during operation by the Control Rod Drive Mechanisms (CRDMs). The Secondary Control Assembly is a secondary reactivity control component in the Natrium Reactor that is used to provide defense in depth relative to common cause failure of absorber bundle to duct binding to address PDC 26 [1].[1] The Secondary Control Assembly is also composed of absorber pins grouped into Control Assemblies, [[

[1] Principal Design Criteria (PDC) 26 specifies the need for independent and diverse means capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the design limits for the fission product barriers are not exceeded.

]](a)(4). More detailed descriptions of the fuel and control assembly designs are provided in Sections 5.2 through 5.4. The Natrium Reactor is currently completing the conceptual design phase therefore some of these details may evolve as more in-depth analysis and testing is performed as part of preliminary design.

5.2 Fuel Assemblies

5.2.1 Fuel Pin

The Natrium fuel pin is comprised of a cladding tube, an upper and lower end cap, wire wrap, sodium-bonded fuel column, fission gas plenum, tag gas capsule, and axial shield (Figure 5-1). The cladding tube and end caps are welded on each end to provide the structural support and hermetic sealing for the contained components.

[[(a)(4)(ECI)]]

Figure 5-1. Natrium Type 1 Fuel Pin

The Natrium Type 1 fuel is metallic uranium alloyed with 10 wt. % zirconium (U-10Zr). The fuel column section of the pin consists of a stack of right circular cylinder fuel slugs. The individual fuel slug lengths are partially influenced by the manufacturer and their optimal process efficiency and capability (within the limits of the fuel specification). The as-manufactured fuel slugs have cross sectional dimensions that represent ~75% of the internal cross-sectional area of the cladding (i.e., 75% smear density). Radiation-induced swelling of the fuel slug will increase its volume such that it contacts the cladding tube inner surface within the first few percent of burnup. The extra space is provided to help ensure interconnected porosity develops in the fuel to promote release of fission gases to the plenum to preclude undue strain on the cladding from fuel-clad mechanical interaction as the fuel continues to generate fission products and swell.

A liquid metal sodium bond is employed in the Natrium Type 1 fuel pin and is initially located in the space between the fuel and cladding. The sodium bond enables adequate heat transfer and prevents unacceptable temperatures during operation, especially at beginning of life when the fuel is not in physical contact with the cladding tube. Once the fuel swells, the liquid metal sodium bond is pushed into the upper plenum although a small amount remains within the porosity of the fuel slugs.

Each fuel pin is helically wrapped with HT9 wire to provide lateral pin-to-pin and pin-to-duct spacing along its length and to promote coolant mixing throughout the assembly. The wire is wrapped under a tensile load. The wire is terminated at each end of the pin [[

]](a)(4)

[[ ]] (a)(4)(ECI)

Figure 5-2. Wire Wrap Fused Ball Termination

The fuel pins have an axial shield section below the fuel column that provides neutron attenuation to limit the damage to the Core Support Structure (CSS) to acceptable levels. The axial shielding is comprised [[ ]](a)(4) located within the sealed pin volume below the fuel column. The shield slug [[

]](a)(4). A fission gas plenum is provided above the fuel and sodium bond to limit internal gas pressure buildup caused by gaseous fission product generation. It is initially backfilled with inert gas.

5.2.2 Fuel Assembly

The fuel assembly is the basic nuclear power generating component of the Natrium Reactor core. It contains the fuel, produces heat, and provides the neutron flux. It can be removed from and replaced or shuffled in the core during reactor refueling. The fuel assembly is principally designed to:

- position the fuel properly in the core for controlled nuclear reaction and generation of thermal power;
- provide passages to guide and control the sodium coolant for heat removal;
- provide shielding to protect components of the CSS from excessive fluence;
- provide features for proper interfacing with other core components, the CSS, the In-Vessel Handling Machines (IVHMs), [[

  ]](a)(4) and,
- provide a physical barrier, the assembly duct, between fuel pins of adjacent fuel assemblies and control assemblies to mitigate or isolate fuel performance impacts on neighboring assemblies

The Natrium fuel assembly is shown in Figure 5-3 and Figure 5-9). It is approximately [[ ]](a)(4)(ECI) total length and comprised of an inlet nozzle, a hexagonal duct tube with above core load pads, a handling socket with top load pads, and a fuel pin bundle with its attachments.

[[ ]](a)(4)(ECI)

Figure 5-3. Natrium Fuel Assembly Design

5.2.2.1 Duct and [[ ]](a)(4)

The hexagonal duct is the principal structural member of the fuel assembly. The fuel assembly duct mates with the handling socket at the top of the assembly, extends the full length, and mates with the inlet nozzle at the bottom of the assembly. The duct tightly encloses the fuel pin bundle along the full length and guides the coolant flow through the bundle, thus permitting an individual assembly orificing scheme that is based on core position. An ACLP is located on the duct approximately two-thirds up from the duct to inlet nozzle connection as shown in Figure 5-3.

The normal duct wall thickness is increased seamlessly at the ACLP for structural support to transmit loads among assemblies and eventually to the core former ring mounted on the in-vessel storage. The ACLP maintains inter-assembly spacing. Between the normal duct thickness and increased thickness at the ACLP, a shallow angle chamfer is provided to minimize withdrawal and insertion loads and to reduce the potential for mechanical binding as the ACLPs move past adjacent fuel assemblies during refueling.

[[

]](a)(4)

[[

]](a)(4)

[[

]](a)(4)

(a)(4)(ECI)

Figure 5-4. Inlet Nozzle with Duct [[ ]](a)(4)

(a)(4)

Figure 5-5. [[ ]](a)(4)

5.2.2.2 Handling Socket

The handling socket, which functions as the TLP in core restraint, is located at the upper end of the fuel assembly and mates with the various grapples during fuel handling operations. The handling socket also guides coolant into the hot pool and into the UIS (for the assemblies under the UIS) during reactor operation and provides spacing and load transfer through hard-face coated load pads (the TLPs) that interface with adjacent core assemblies and the core former ring of the CRS. [[

]](a)(4). Fuel assemblies may require rotation at selected irradiation times to balance out the irradiation-induced geometrical distortions, and the orientation notch on the handling socket collar provides the reference to achieve the desired amount of rotation.

(a)(4) (ECI)

Figure 5-6. Handling Socket [[ ]](a)(4)

5.2.2.3 Inlet Nozzle and Pin Attachment Hardware

The inlet nozzle, located at the lower end of the fuel assembly, interfaces with, and provides the primary load path of the fuel assembly to the CSS. It provides the coolant inlet flow path to the assembly internals and contains orifice plates to modify the total flow within the assembly based on the location within the core. The inlet nozzle also interacts with the receptacle to create a hydraulic hold down force on the assembly from the pressure difference between the inlet plenum and low-pressure plenum. [[

[1] Specific notching scheme shown in figure is an example for illustrative purposes.

]](a)(4)

[[ ]] (a)(4) (ECI)

Figure 5-7. Inlet Nozzle [[ ]](a)(4) Cross Section

Assembly bypass coolant flow is minimized [[

]](a)(4)

[[

]](a)(4)

Primary vertical loads on the assembly are carried through the nozzle, then transferred to the CSS receptacle. Horizontal loads are transferred to the core structure in two locations: at the upper section of the nozzle, horizontal loads are transferred to the upper grid plate radially through a centering collar; in the lower section, mating features in the nozzle transfer lateral loads to the receptacle cylinder, which carries the load through a moment couple to the lower grid plate. The inlet nozzle has a conical transition from the overall assembly hexagonal shape to the nozzle circular shape. This conical feature provides self-alignment capability during fuel assembly handling operations. The nozzle has machined slot features located at the top that mate with the pin attachment hardware [[ ]](a)(4). In turn, the pin attachment hardware mates with the lower end cap of each fuel pin and provides axial restraint and support (Figure 5-8).

The fuel assembly in general, and the inlet nozzle in particular, have design features that permit liquid sodium to drain from all internal volumes to minimize sodium residuals when removed from the core and ex-vessel storage prior to Post-Irradiation Examination (PIE) shipment or storage as spent fuel. The fuel assembly is also designed to be washed in the Sodium Removal System to remove all residuals prior to placement in long-term spent fuel storage.

[[ ]](a)(4) (ECI)

Figure 5-8. Fuel Pin Attachment Hardware for a [[ ]](a)(4)(ECI) Pin Bundle

5.2.2.4 Fuel Pin Bundle

The fuel pin bundle is shown in Figure 5-9 with its hexagonal cross section. Each fuel assembly contains [[ ]](a)(4)(ECI) sealed fuel pins packed with triangular pitch spacing. The pins extend from their primary attachment point near the top of the inlet nozzle to just below the handling socket at the top of the assembly. The bundle is comprised of [[ ]](a)(4) separate strip layers varying from [[ ]](a)(4) pins per layer (see Figure 5-8). The upper and lower end caps of the wire-wrapped fuel pins in all of the strip layers have the same orientation to ensure uniform coolant flow across the bundle and proper fit with all attachment hardware. A very tight fit of the fuel pin bundle in the duct is important to achieve proper coolant flow so that the coolant is guided to the bundle internals and a minimum amount of coolant flows through the bypass region between the bundle periphery and inner duct wall. Accordingly, the bundle is fixtured and slightly compressed during assembly prior to installing the duct over the bundle, targeting a maximum bundle clearance to the inner duct on the order of the thickness of [[ ]](a)(4).

(a)(4) (ECI)

Figure 5-9. Natrium [[ ]](a)(4)(ECI) Fuel Pin Bundle Cross Section in Fueled Region[1]

5.2.3 Lead Demonstration Assembly

In an effort to mitigate licensing risks, the Type 1 fuel design has leveraged historic fuel designs (e.g., FFTF and PRISM Mod B) and operating targets as much as practical. This enables application of historic fuel operating experience to support fuel qualification. Moreover, a fuel surveillance program within the Natrium plant will be established to address any potential gaps in the fuel qualification program that are not adequately covered by historic experience or readily addressed with new test data.

Given the establishment of this fuel surveillance program, the core will have the capability to irradiate fueled Lead Demonstration Assemblies (LDAs) that support rapid post-irradiation exams of fuel pins. These LDAs will be designed to have unique features and components that allow for remote disassembly and removal of a select number of irradiated fuel pins. Following removal from the assembly, the selected fuel pins will then be examined to gather data to reduce fuel performance uncertainties and potentially increase fuel burnup targets. The LDA will be designed to be very similar to that of the standard Type 1 fuel assembly. It will have the same assembly height, employ a subset of the same standard fuel pins, fit within the same hexagonal grid as the standard fuel assembly, and the structural members and external configuration of the LDA will be largely identical to those of the fuel assemblies. Additionally, LDAs will be designed to be hydraulically and thermally compatible with the other fuel assemblies, i.e., to have compatible

[1] The duct wall is shown crosshatched and the ACLP is revealed at the outer edge due to its larger diameter

pressure drop and to meet the cladding and fuel temperature limits. [[

]]$^{(a)(4)}$

The unique aspects of the LDAs involve a select number of positions in the fuel bundle that contain removable pins and the associated pin retention components. The LDAs will require unique components to enable remote disassembly for extraction of the removable pins. [[

]]$^{(a)(4)}$

5.2.3.1 LDA Handling Socket

Traceability of the position of the removable pins within the LDA bundle will be maintained at all times to ensure the appropriate pins are removed to support post-irradiation exams. To aid in locating the appropriate pins, the handling socket has features that provide orientation relative to the mapping of the removable pins. Initial orientation of the LDA assemblies will be established and recorded during core assembly fabrication, whereby the duct and handling socket will be installed with a prescribed orientation relative to the removable pin mapping. [[

]]$^{(a)(4)}$

5.2.3.2 LDA Fuel Pin Bundle

The LDA fuel bundles consist of an array of wire wrapped and non-wire wrapped fuel pins arranged in a tight triangular pitch spacing. As many as [[ ]]$^{(a)(4)}$ removable pins will be designated in each LDA for pin removal and examination. [[

]]$^{(a)(4)}$

An illustrative example of the location of the removable pins in the assembly is shown in Figure 5-10. The final selected positions will be determined as the fuel surveillance program is further defined.

[[ ]] (a)(4)

Figure 5-10. Potential Removable Pin Locations

5.2.3.3 LDA Removable Fuel Pin and Retention Component

The LDA fuel pins extend from their primary attachment point near the top of the inlet nozzle to just below the handling socket. They will extend beyond the rest of the standard pin bundle with a feature on the upper end cap to permit positive gripping with the pin removal tool.

Vertical retention of the pin is provided by a device at the lower end that interfaces with the pin rails and connects to the special lower end cap of the removable pin. [[

]](a)(4)

Excluding the special design of [[ ]](a)(4) the removable pins are otherwise identical to the standard fuel pins, i.e., they are fabricated with the same cladding, fuel, sodium bond, axial shield, and plenum as the Type 1 fuel pins, using the same manufacturing specifications. Even though the lower end of the pin is different than a standard fuel pin due to the special retention device, the fuel column axial elevation will be kept level with the rest of the fuel.

5.2.3.4 LDA Secondary Pin Restraint

A secondary pin restraint device is incorporated into the LDAs to prevent ejection of any removable pin that has inadvertently lost primary restraint from its retaining socket during reactor operation. The secondary restraint feature is designed to facilitate remote removal during LDA disassembly.

5.2.4 Lead Test Assemblies / Type 1B Fuel

The Natrium core has the capability to irradiate fueled Lead Test Assemblies (LTAs). These lead test assemblies have innovative features that allow them to achieve long reactor residence times and high burnup and higher coolant outlet temperatures to improve fuel cycle economics. The LTA program plan for the Natrium core is still under development; however, [[

]][(a)(4)] provides a conceptual LTA plan along with additional testing and analysis work that will be used to support Type 1B fuel qualification [13]. The following discussion provides current concepts being considered for the LTA program that may change depending on innovations, advancements or other information that becomes available over time.

The LTA is very similar to the lead demonstration assembly. It has the same assembly height, fits within the same hexagonal grid as the LDA, and the structural members and external configuration of the LTA are identical to those of the fuel assemblies (i.e., same inlet nozzle, handling socket, and duct mechanical joint). Additionally, LTAs are designed to be hydraulically and thermally compatible with standard fuel assemblies, i.e., to have the same pressure drop and to meet the cladding and fuel temperature limits. The unique aspects of the LTA are the LTA fuel pin and the duct material.

5.2.4.1 LTA Duct

The LTAs will utilize the same assembly pitch as is employed for the other core assemblies. They will use [[

]][(a)(4)] in Reference [13].

5.2.4.2 LTA Fuel Pin Bundle

The LTA fuel bundle consists of an array of wire-wrapped fuel pins arranged in a tight triangular pitch spacing. Similar to the Type 1 fuel pin bundle, the wire wraps of all the pins in the bundle are oriented exactly the same to ensure uniform coolant flow across the bundle and proper packing and fit-up with all attachment hardware. Like the LDA, [[
]][(a)(4)] support rapid removal of the pins for subsequent PIE.

5.2.4.3 LTA Fuel Pin

The key differences between the LTA fuel pins (Type 1B) and the host core sodium bonded fuel pins (Type 1) are shown in Figure 5-11. [[

]][(a)(4)]

(a)(4)(ECI)

Figure 5-11. Key Differences of Type 1 and Type 1B (LTA) Fuel Pin Design [1]

5.3 Control Assemblies

The Reactivity Control System is the principal nuclear control system of the Natrium Reactor core and its main function is to position neutron absorber material to appropriately control and terminate the nuclear reaction. This function is consistent with the system requirements for providing safe and predictable operation of the reactor. This system meets reactor shutdown requirements without the aid of any other reactivity control system. A total of [[ ]](a)(4) penetrations are provided in the reactor head directly above those core locations, one for each reactivity control unit. The Control Rod Drive Mechanism (CRDM), Control Rod (CR) driveline (CRD), and control assembly are directly coupled during normal operation.

The control rods (located within their own dedicated hexagonal control assemblies in the core) are driven by the CRDM to move and position absorber material vertically within the core to control core reactivity and power to maintain fuel within acceptable design limits. They have the capability to control core reactivity changes during expected operations and specified accident conditions with variations in core composition during operation over the life of the core. The control rods also provide power response for the plant control and data system (PCD). Finally, control rods provide SCRAM insertion capability with sufficient reactivity worth to shut down the reactor and maintain it in cold shutdown even if the highest worth rod is stuck in the withdrawn position. To provide design diversity

[1] This image is only illustrative with some design details omitted to simplify comparison.

to limit common cause failures there are both primary and secondary control assemblies, [[ ]](a)(4).

5.3.1 Absorber Pin

The control rod absorber pin is a sealed, helium bonded design that is comprised of a cladding tube, upper and lower end cap, wire wrap, boron carbide pellet column, gas plenum, spring, and plenum spacer. The cladding tube is made of HT9. The cladding tube is hermetically sealed with end caps that are welded on each end and provide the structural supports for the pellet, spring, and spacer components.

Natural boron carbide is used as the absorber material in the pin and is shaped into the form of pellets that are manufactured by pressing and sintering powder into right circular cylinders. The selection of boron carbide as the baseline absorber material is based on its successful and long-standing use in fast reactors around the world, including the extensive irradiation program conducted at the FFTF, and the availability of irradiation data for licensing and qualification. Compared to other absorber materials, boron carbide has advantages due to its relatively high neutron absorption cross-section, availability and low cost, comparative ease of fabrication, and low radioactivity after irradiation. The pellet-to-clad gap is provided to accommodate pellet swelling that may limit the lifetime of the pin due to strain in the cladding.

The pins have an upper plenum to accommodate the gaseous fission products released from the B_4C absorber column during irradiation. The plenum volume is sized such that cladding stresses and strains due to internal gas pressure are maintained to acceptable levels throughout life. The plenum also accommodates any axial absorber column growth. A spring and plenum spacer is provided to ensure that the B_4C column maintains its axial position during preoperational shipping and handling and permits axial expansion during operation.

Each of the CR pins is helically wrapped with a wire to provide lateral pin-to-pin and pin-to-duct spacing along its length and to promote coolant mixing throughout the CR. The wire is wrapped under a tensile load. Like the fuel pins, the wire is terminated at each end of the pin [[ ]](a)(4).

5.3.2 Primary Control Rod Assembly

The primary control rods employ [[ ]](a)(4) absorber pins that are arranged in a triangular pitch, packed tightly into a hexagonal lattice, and surrounded by a [[ ]](a)(4) HT9 duct (the control rod duct) as shown in Figure 5-12. The control rod duct is the principal structural member for the absorber pin bundle between the upper and lower guide plates.

The CR is designed to move freely within the control assembly duct, with its own dedicated space within the reactor core, throughout its design lifetime. Speed of the control rod insertion in a SCRAM is maintained throughout life accounting for the worst-case distortions, including bowing, misalignment, and friction between the inner and outer duct. The CR duct is welded to the upper and lower guide plates, and to the coupling head that connects the pin bundle to the CRD. Interface wear pads are provided at the top and bottom of the CR at the plate locations to provide a smooth gliding surface against the inner surface of the assembly duct. These wear pads are hard coated to minimize friction and the potential for galling with the control assembly duct. A sufficiently sized gap is provided between the wear pads and the control assembly duct to allow for free motion under all conditions and to accommodate anticipated distortions. Three-point contact along the vertical length of the control assembly duct is precluded by proper design of the interface between the CR and assembly duct, and specifically the CR wear pads and associated gap, assuming worst case distortions. Control assemblies will also be rotated during normal refueling outages to reverse assembly bowing deformations and extend the assembly operational lifetime. The gap size between the inner and outer duct has additional design requirements such that thermal-hydraulic considerations are satisfied (e.g., bypass flow around absorber bundle) and potential reactivity oscillations due to flow induced lateral motion of the control rod are minimized to acceptable levels. Control assemblies are replaced in the core after their design lifetime is achieved during normal reactor refueling. The gap between the inner and outer control ducts can be seen in Figure 5-13.

(a)(4)(ECI)

Figure 5-12. Illustrative Example of a Natrium Control Assembly

The absorber pins are attached at the top of the control rod on pin rails that connect via support bars to the upper guide plate, as shown by Figure 5-14. [[

]](a)(4)(ECI) By design, and as a requirement, the CR always decreases core reactivity when inserted incrementally into the core, even accounting for the effects of absorber material depletion over its design lifetime. At the fully inserted position, or after a SCRAM is accomplished, the control rod B_4C column is aligned with the fuel column at their respective centerlines. [[

]](a)(4)(ECI)

[[ ]](a)(4)(ECI)

Figure 5-13. Cross Section View of Natrium Control Assembly

[ ] (a)(4)(ECI)

Figure 5-14. Natrium Control Rod Absorber Pin Attachment

[ ] (a)(4)(ECI)

Figure 5-15. Control Assembly – Damper and Driveline Assemblies

5.3.2.1 Duct, ACLP, Handling Socket, TLP, and [[ ]](a)(4)

The control assemblies utilize the same HT9 duct, ACLP, handling socket, TLP, and [[ ]](a)(4) as described for the fuel assembly in Section 5.2.2.

5.3.2.2 Inlet Nozzle and Axial Shield Block

The control assembly inlet nozzle is similar in design to the fuel assembly inlet nozzle except that it does not have features for interfacing with any pin attachment hardware. Instead, it is connected to an axial shield block that provides neutron attenuation to limit the damage to the CSS. The shield block is housed in each control assembly duct located directly above the inlet nozzle. It has machined through-holes to permit the flow of coolant from the inlet nozzle to the internals of the control assembly (Figure 5-16).

(a)(4)(ECI)

Figure 5-16. Control Assembly Lower Detail

5.3.2.3 Control Rod Connection to the Control Rod Drive

The CRs are connected and disconnected to the CRD (Figure 5-17) at the coupling head so that the associated drivelines may be lifted above the reactor core assemblies to permit plug rotation. This disconnect point is also used during a SCRAM so that the CRs drop without connection to the CRD. The control assemblies with the CRs inside are removed from the core by the IVHM in the same manner as fuel assemblies. [[

]](a)(4)

[[ ]](a)(4)(ECI)

Figure 5-17. Natrium Control Rod Connection

5.3.3 Secondary Control Rod Assembly

The Secondary Control Rod is still under development to ensure that the geometric differences between the Primary and Secondary control rods are sufficient to meet diversity requirements. While the Primary Control Rod design is expected to meet the licensing requirements for reliability in reactivity control, the Secondary Control Rod design is intended to create geometric diversity to preclude common cause failures that would inhibit SCRAM or control functionality by mechanical binding phenomena. While there are different forms of mechanical binding, the diversity requirement for the control rods focuses on inner to outer control rod duct interactions that could lead to significantly increased SCRAM time, making the transient impact more severe, or the potential of mechanical binding that would stop control rod insertion (jamming phenomenon).

Currently, the Secondary Control Rod Assembly is anticipated to be nearly identical to the Primary Control rod assembly as outlined in Section 5.3.2. This includes descriptions of pin geometry, assembly configurations, coatings descriptions, etc. The key differences between the Primary and Secondary Control Assemblies are only in the number of absorber pins used, changes to the control assembly geometry, and the space between the inner duct and the guide tube as shown in Figure 5-18. The figure shows the difference between the [[ ]](a)(4)(ECI) pin assembly and a [[ ]](a)(4)(ECI) pin bundle used in the Secondary Control Rod. The change between the two can be seen clearly by removing the [[ ]](a)(4)(ECI). This change opens a significant gap (shown in white) between the guide tube and the inner hex-duct for the control rod. The connection gaps at the dashpot / tie plate at the top of the bundle remain the same. The change to the pin bundle geometry, however, is expected to be significant enough to preclude the previously mentioned common cause mechanical binding failure that could be experienced by either the Primary or Secondary Control Rods as they move within their own dedicated assemblies. The additional space between the inner and outer duct should also limit the likelihood or amount of force for any inner and outer duct friction contact.

(a)(4)(ECI)

Figure 5-18. Primary vs. Secondary Control Rod Bundle Cross-Section

5.4 Core Restraint System

The Core Restraint System (CRS) is an important interface consideration in the design of all core assembly types including both fuel and control assemblies. One of the primary functions of this system is to control radial expansion reactivity feedback that results from assembly displacements within an SFR core. Changes in thermal, irradiation, and mechanical loads at different state-points of reactor operation induce core-wide assembly movement that can cause reactivity changes affecting power. The mechanical (and consequently reactivity) response of the core restraint system is heavily dependent on aspects of core assembly design. As such, a system overview, relevant phenomena, assembly design dependencies, and design targets are described in this section.

The core restraint system consists of core former rings connected to the CSS, assembly top load pads (TLPs), assembly above core load pads (ACLPs), assembly inlet nozzles, and receptacles at the bottom of the CSS. A schematic of these components is shown in Figure 5-19 [14]. The assembly inlet nozzle is inserted into the CSS receptacle, which directs vertical loads to the CSS.

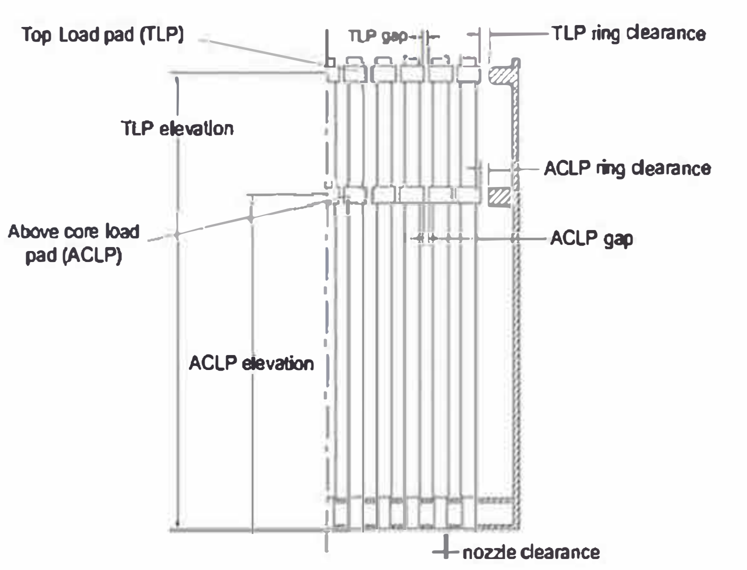

Figure 5-19. Schematic of a Core Restraint System and Impactful Parameters

Core assemblies are subjected to temperature and neutron flux distributions over their residence time. These distributions induce bowing along the axial height of the assembly. This occurs due to differential expansion of opposing sides of the assembly duct component when subjected to a temperature gradient or fluence gradient distribution. For example, an assembly with a temperature gradient across its cross section will deform or bow toward the colder side as shown in Figure 5-20. Additionally, assembly load pads will interact with neighboring assembly load pads or the CSS, which will impose bending deformations on the assembly and cause rotations of the inlet nozzle within the receptacle. This is also illustrated in Figure 5-20 as a restrained thermal bowing case with an example resultant deformation shown. These restraint loads at the assembly ACLP and TLP are imparted by neighboring assemblies and the core support structure, which together comprise the core restraint system as previously described.

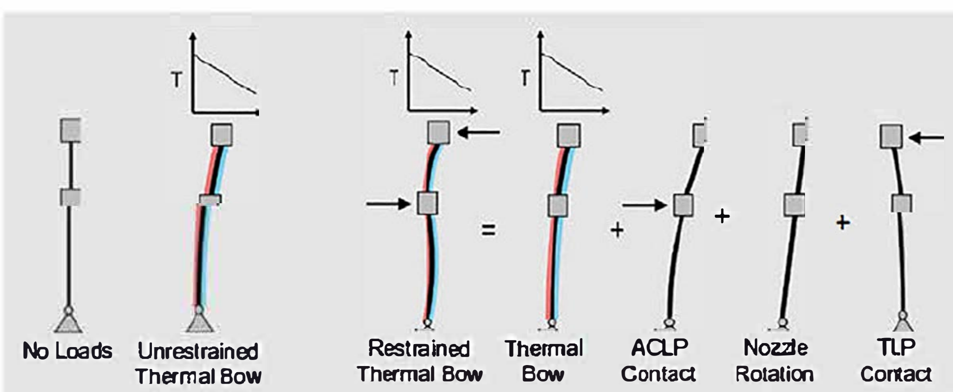

Figure 5-20. Contributing Effects to Core Assembly Restrained Thermal Bowing Deformations

Nonlinear material effects are an additional important consideration in the design of the core restraint system and the core management program. Over the residence time of an assembly at high temperature, high fluence, and bending loads, assembly structural materials will undergo inelastic strains including thermal creep, irradiation creep, and void swelling. A properly designed core restraint system will account for residual assembly deformations and ensure deformed assemblies can be removed from the core so unirradiated assemblies can be introduced. An example illustrating the importance of this consideration is the EBR-II core which was designed prior to an understanding of void swelling in core structural materials. EBR-II experienced significant refueling challenges due to these inelastic deformations at assembly contact locations, which became highly limiting to assembly residence time. FFTF also experienced refueling challenges with high-deformation assemblies requiring more complex refueling operations to retrieve.

Fast reactor cores are highly sensitive to fuel motion that occurs primarily due to core assembly bowing. Other events such as seismic loading can also induce fuel motion. Core assembly deformations influence the neutron balance within the core (primarily neutron leakage), which lead to changes in reactivity. This reactivity feedback is significant in SFRs and is understood to be a primary contributor to the core melt accident at EBR-I. As such, reactivity feedback induced by the core restraint system plays a role in reactor stability and is a consideration in reactor safety analysis.

In order to achieve a determinate and predictable configuration of core assemblies, fully developed load paths need to develop between various core assembly ACLPs and TLPs, eventually reacting radially at the CSS. This condition is referred to as a “mechanically locked” core and is achieved when core-wide inter-assembly gaps are sufficiently closed, primarily through core assembly bowing, and adequate inter-assembly contact is established. Once sufficient contact and a mechanically locked core condition is established, the core restraint system tends to insert negative reactivity with increases in core power.

The core restraint system has competing requirements that relate primarily to inter-assembly gap management. Designing smaller inter-assembly gaps generally produces more favorable reactivity behavior both from assembly bowing and seismic events. With smaller gaps between individual assemblies, a mechanically locked core condition can typically be achieved earlier as there are less cumulative core-wide assembly gaps that need to be overcome by core assembly bowing to establish sufficient core-wide inter-assembly contact. This indicates larger ranges of operation within the mechanically locked core condition, which in-turn provides more stable reactivity feedback behavior. Additionally with smaller inter-assembly gaps, potential reactivity insertions from core restraint are generally smaller in magnitude from both bowing and seismic loading as there is less space for assemblies to translate before core-wide assembly contact is established. Tighter inter-assembly gaps also promote better core assembly alignment for interfacing systems such as the IVHM or control rod driveline.

The competing design requirement strives to maintain sufficiently large inter-assembly gaps for ease of core management. High residence fuel assemblies experience significant inelastic material deformations due to extended time at high temperatures, fluences, and bending loads applied by the core restraint system. During refueling operations, high deformation assemblies can cause excessive handling loads on the IVTM if the amount of assembly deformation is excessive relative to the inter-assembly gaps available through which to withdraw the assembly. Depending on core assembly material selection, different mechanisms can dominate inelastic material deformation. Austenitic stainless steels, such as 316 SS, exhibit swelling-dominated deformation behavior while ferritic-martensitic stainless steels, such as HT9, exhibit creep-dominated deformation behavior. Designing sufficiently large inter-assembly gaps allows for removal of high deformation assemblies from the core without exceeding the load capacity of the handling machine.

5.5 Materials

5.5.1 HT9

Alloy HT9 stainless steel, referred to here as HT9, is the material selected for fuel cladding, ducts, and other fuel components. Because austenitic stainless steels undergo excessive radiation-induced void swelling prior to reaching the desired fuel life, fast-reactor cladding and duct development programs of the past switched focus from austenitic to ferritic or ferritic-martensitic (FM) steels for long exposure applications. Selected FM steels, including HT9, exhibit strong resistance to swelling, maintain adequate microstructural stability under irradiation, and retain adequate ductility at typical reactor operating temperatures.

HT9 and analogous alloys, like the Russian EP-823, have been considered top candidate materials for nuclear reactor core components since the Liquid Metal Fast Breeder Reactor (LMFBR) era in the 1970s and 1980s. In the US fast-reactor development program, HT9 developed by Sandvik was selected for cladding and ducts for the Integral Fast Reactor (IFR) concept [15], as the next generation fuel cladding for the EBR-II [16], and as the cladding and duct material for the metallic fuel assemblies in the FFTF [17]. For the Natrium Reactor, the pathway for qualification of HT9 as fuel component material is centered on leveraging test data and component operating experience from these programs. Additionally, TerraPower has invested substantially in developing HT9 material more recently and plans to demonstrate any improvements over legacy material as described in Section 6.3 of this document.

The scope of qualification of HT9 for the Natrum Reactor is currently limited to the fuel system. The fuel system consists of the fuel assemblies, reflector assemblies, shield assemblies, and reactivity control assemblies.

Component designs using HT9 material in the fuel system are cladding, ducts, upper and lower fuel pin endcaps, and wire wrap.

5.5.1.1 Composition

Alloy HT9 steel is a FM Cr-Mo stainless steel whose evolution and composition can be traced to AISI 430 and AISI 410, the basic general-purpose alloys of the ferritic stainless-steel family. FM stainless steels are generally defined as those containing at least 9 wt.% chromium and have microstructures of α-iron (ferrite), martensite, and carbides. Several standard and nonstandard alloy types have been derived from the basic alloy by varying the composition to achieve specific properties. Typical alloying elements in addition to Cr and Mo are W (up to 3 wt.%), V (< 0.5 wt.%), and Nb (< 0.5 wt.%). Alloy HT9 is classed as a 12Cr-MoVW type. The W addition endows this grade with greater strength than corresponding steels without W.

The nominal composition of HT9 is in the table below, taken from Chapter 18 of the Fuel Cycle Research and Development (FCRD) Materials Handbook [18].

Table 5-1. Nominal Composition of HT9 Steel

Element	Weight (%)
[[	
	]] (a)(4),(ECI)

The product forms of HT9 needed for fuel components are cladding tubes (fuel and absorber), duct tubes, wire, bar (for endcaps), and sheets (for welded ducts, if required).

Qualification of HT9 must include evaluation of welds. Resistance pressure welding (RPW) is being developed for fuel pin end caps, while other forms of welding are being considered for ducts if required.

5.5.1.2 Manufacturing Process

More than eight metric tons of HT9 have been manufactured to date in 18 heats during development by TerraPower, supporting the definition of manufacturing steps as listed in the table below.

Table 5-2. Manufacturing Steps for HT9 Components

Step	Critical Parameters
Melt	[[ ]](a)(4)
Homogenization	[[ ]](a)(4)
Forge	[[ ]](a)(4)
Hot Work (Extrusion / Rolling)	[[ ]](a)(4)
Cold Work (Drawing / Rolling) and Intermediate Anneal	[[ ]](a)(4)
Final Heat Treatment	[[ ]](a)(4)

5.5.2 U-10Zr Fuel

As described above, metallic uranium alloyed with 10wt% zirconium (U-10Zr) is the fissile material used as the fuel slugs in Natrium Type 1 fuel pins. The enrichment of the uranium is determined by the position within the Natrium core, with the peak enrichment being <20% U-235. [[

]]$^{(a)(4)}$

5.5.2.1 Manufacturing Process

Historically, metallic fuel slugs have typically been manufactured via an injection casting process where the uranium and zirconium (or other fuel alloy components) are loaded into a crucible and melted via induction heating. After the fuel alloy components have melted and mixed, a vacuum is drawn, and an array of quartz molds suspended above the fuel melt is lowered partially into the melt. Pressure is then rapidly applied by introducing inert gas inside of the casting furnace driving the liquid fuel alloy up into the quartz molds, due to differential pressure, where it quickly solidifies. The molds are removed from the crucible and the system allowed to cool. The molds are removed from the injection casting furnace and broken away to reveal the fuel slugs. The fuel slugs are sheared to length and inspected prior to incorporation in fuel pins.

TerraPower plans on relying heavily on the irradiated metallic fuel data from EBR-II and FFTF, which was fabricated with injection casting. To mitigate potential risks of unanticipated performance introduced by alternate fabrication processes, TerraPower will use the injection casting process for Type 1 fuel slug fabrication. TerraPower is relying heavily on knowledge transfer from the DOE labs in developing its new injection casting capabilities, including historic system drawings, casting system design specifications, operating procedures, fuel specifications, and interviews with former operators.

TerraPower previously performed detailed characterization of archived fresh U-10Zr fuel materials from EBR-II and FFTF/MFF [19] [20]. When newly manufactured U-10Zr slugs are available from prototype equipment testing characterization will be performed to verify consistency with legacy materials.

5.5.3 Other Core Materials

For some of the components of the fuel system other materials are being considered. This includes 304 and 316 austenitic stainless steel, and Inconel 718 nickel-based superalloy. These materials are well established and are widely used in the nuclear industry, with extensively documented performance and properties. Design inputs taken from NRC accepted standards for these types of materials are considered pre-qualified.

5.6 Verification of the Fuel System Design Basis

Fuel system descriptions and design drawings are required to support safety analyses to provide the information necessary to verify that the fuel system design bases are met. The specific details adapted from the Standard Review Plan [2] are summarized in Table 5-3, along with the associated documentation/media planned to provide the required fuel design information.

Table 5-3. Summary of Completed and Planned Activities to Satisfy Fuel System Design Description Requirements (RAC 4.2-5)[1]

Expected Format of Information	Required Fuel System Information with Associated Tolerances	Sample Reference or Future Activity to Address
Design Description of Cladding for Fuel/Absorber Pins	Type and metallurgical state of the cladding	[[ ]]$^{(a)(4)}$
	Cladding outside diameter	[[ ]]$^{(a)(4)}$
	Cladding inside diameter	[[ ]]$^{(a)(4)}$
	Cladding roughness	[[ ]]$^{(a)(4)}$
Design Description of Fuel Pin	Fuel slug density	[[ ]]$^{(a)(4)}$
	Fuel slug diameter	[[ ]]$^{(a)(4)}$
	Fuel slug length	[[ ]]$^{(a)(4)}$
	Fuel slug grain size	[[ ]]$^{(a)(4)}$
	Slug alloy composition for metallic fuel	[[ ]]$^{(a)(4)}$
	Allowable slug impurities	[[ ]]$^{(a)(4)}$
	Shield slug parameters	[[ ]]$^{(a)(4)}$
	Sodium bond height	[[ ]]$^{(a)(4)}$
	Fuel column length	[[ ]]$^{(a)(4)}$
	Overall pin length	[[ ]]$^{(a)(4)}$
	Fill gas type and pressure	[[ ]]$^{(a)(4)}$
	End plug dimensions	[[ ]]$^{(a)(4)}$
	Wire wrapping dimensions	[[ ]]$^{(a)(4)}$
	Fissile enrichment and isotopics	[[ ]]$^{(a)(4)}$
Design Description of Absorber Pin	Pellet density	[[ ]]$^{(a)(4)}$
	Pellet diameter	[[ ]]$^{(a)(4)}$
	Slug grain size	[[ ]]$^{(a)(4)}$
	Pellet chemical composition for absorber	[[ ]]$^{(a)(4)}$
	Allowable pellet impurities	[[ ]]$^{(a)(4)}$
	Shield slug parameters	[[ ]]$^{(a)(4)}$
	Plenum height	[[ ]]$^{(a)(4)}$
	Plenum spring	[[ ]]$^{(a)(4)}$
	Absorber column length	[[ ]]$^{(a)(4)}$
	Overall pin length	[[ ]]$^{(a)(4)}$
	Fill gas type and pressure	[[ ]]$^{(a)(4)}$
	Upper end plug dimensions	[[ ]]$^{(a)(4)}$
	Lower end plug dimensions	[[ ]]$^{(a)(4)}$
	Wire wrapping dimensions	[[ ]]$^{(a)(4)}$
	Boron enrichment and isotopics	[[ ]]$^{(a)(4)}$
Design Drawings	Fuel assembly cross section	[[ ]]$^{(a)(4)}$
	Fuel assembly outline	[[ ]]$^{(a)(4)}$
	Fuel pin schematic	[[ ]]$^{(a)(4)}$

[1] Additional guidance related to required description of the fuel system is provided in Regulatory Guide 1.206 – Combined License Applications for Nuclear Power Plants.

Expected Format of Information	Required Fuel System Information with Associated Tolerances	Sample Reference or Future Activity to Address
	Fuel pin wire wrap location	[[ ]](a)(4)
	Absorber pin schematic	[[ ]](a)(4)
	Wire wrap location	[[ ]](a)(4)
	Inlet and outlet nozzles	[[ ]](a)(4)
	Control rod duct with respect to control rod dimensions	[[ ]](a)(4)
	Control rod assembly cross section	[[ ]](a)(4)
	Control rod assembly outline	[[ ]](a)(4)
	Control rod schematic	[[ ]](a)(4)

6. FUEL SYSTEM DESIGN EVALUATION

The Natrium Reactor will use analytical predictions to evaluate fuel system compliance with the design basis limits, while pointing to operating experience of similar historic metallic fuel pin designs and relying on testing to help bridge the gap between historic experience/designs and Natrium fuel design parameters. Analytical predictions of Natrium Reactor performance will evaluate the fuel system design for physically feasible combinations of chemical, thermal, irradiation, mechanical, and hydraulic interactions. The evaluation of these interactions will include the effects of normal operations, AOOs, and LBEs [39]. New tests will largely focus on understanding high-importance phenomena needed to evaluate compliance to the fuel design bases. To aid in the identification of which RAC are addressed by fuel system design limits, Table 4-1 through Table 4-4 summarize the correspondence of fuel system design criteria with the RAC for fuel damage, failure, coolability, and control rod insertability.

To aid in the identification of high-importance phenomena, a Phenomena Identification Ranking Table (PIRT) analysis was performed evaluating the applicable phenomena for each fuel and absorber design criteria [40] [41] [42]. These assessments were performed by convening a team of experts within TerraPower with representatives from the Fuels, Materials, Safety, and Mechanical teams to assess the applicable phenomena for each fuel pin design limit and the relative Importance and Knowledge Level of the respective phenomena. The internal definitions used for determining the Importance rankings and Knowledge Levels are summarized in Table 6-1 and Table 6-2, respectively.

Table 6-1. Importance Ranking Definitions

Importance Ranking	Definition
Low (L)	Small influence on demonstrating compliance ± 1σ variation of parameter/phenomenon has minimal impact on prediction of design criterion
Medium (M)	Moderate influence on demonstrating compliance ± 1σ variation of parameter/phenomenon has moderate impact on prediction of design criterion
High (H)	Significant influence on demonstrating compliance ± 1σ 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% to 100% of complete knowledge and understanding
Partially Known (P)	30% to 70% of complete knowledge and understanding
Unknown (U)	0% to 30% of complete knowledge and understanding

6.1 High-importance Phenomena

The PIRT assessments were performed to help identify the high-importance phenomena that must be accounted for when evaluating the performance of Type 1 fuel and control assemblies. The high-importance phenomena for fuel and absorber pins are summarized in Table 6-3, along with the corresponding design criteria and associated RAC. When evaluating and consolidating the high-importance phenomena, it became clear that many of the identified phenomena are more aptly described as operating parameters/conditions (e.g., Cladding Temperature, Fuel Burnup) versus complex physical phenomena (e.g., Fission Gas Release or FCCI), so these different categories were also noted in Table 6-3. To help prioritize activities, an “Overall Knowledge Level” ranking is also included in Table 6-3, which is the average Knowledge Level determined for each identified high-importance phenomena/parameter. Table 6-4 is a summary of high-importance phenomena for fuel and control assemblies.

Table 6-3. Summary of Identified High-Importance Phenomena and Associated Design Limits and RAC for Fuel and Absorber Pins

Category	High-importance Phenomena/Parameters	Overall Knowledge Level	Applicable Design Limit	Applicable RAC
Fuel Pin Phenomena	Fission gas release	P	Total Peak Cladding Strain, Peak Cladding Thermal Creep	4.2-1.1, 4.2-1.6, 4.2-1.8, 4.2-2.3, 4.2-2.5, 4.2-3.1, 4.2-3.3, 4.2-3.5
	HT9 mechanical response as a function of temperature, stress, irradiation, and time	K	Total Peak Cladding Strain, Fatigue Limit, Peak Cladding Thermal Creep Strain	4.2-1.1, 4.2-1.6, 4.2-1.8, 4.2-2.3, 4.2-2.5, 4.2-3.1, 4.2-3.3, 4.2-3.4, 4.2-3.5
	FCCI	P	Cladding Wastage, Peak Cladding Thermal Creep Strain	4.2-1.3, 4.2-1.4, 4.2-1.5, 4.2-2.3, 4.2-2.5
	Fuel Thermal Conductivity	P	Peak Fuel Temperature	4.2-2.2,4.2-3.4
Absorber Pin Phenomena	Gas release	K	Total Peak Cladding Strain	4.2-1.1, 4.2-1.7, 4.2-1.8, 4.2-1.11,4.2-4.1
	HT9 mechanical response as a function of temperature, stress, irradiation, and time	P	Total Peak Cladding Strain, Fatigue Limit	4.2-1.1, 4.2-1.2, 4.2-1.7, 4.2-1.8, 4.2-1.11, 4.2-4.1
	ACCI	U	Total Peak Cladding Strain, Total Peak Cladding Wastage,	4.2-1.3, 4.2-1.4, 4.2-1.5, 4.2-1.11, 4.2-4.1
	B_4C Swelling/ACMI	K	Total Peak Cladding Strain, Peak Absorber Temperature	4.2-1.1, 4.2-1.7, 4.2-1.8, 4.2-1.11, 4.2-4.1
	Absorber cracking/fragmentation	P	Peak Absorber Temperature Limit	4.2-1.11
Fuel Pin Parameters/ Operating Conditions	Fuel Burnup	P	Total Peak Cladding Strain, Fatigue Limit, Peak Cladding Thermal Creep	4.2-1.1, 4.2-1.6, 4.2-1.8, 4.2-2.3, 4.2-2.5, 4.2-3.1, 4.2-3.3, 4.2-3.5
	DPA on cladding	P	Total Peak Cladding Strain, Fatigue Limit	4.2-1.1, 4.2-1.6, 4.2-1.8
	Cladding temperatures	P	Total Peak Cladding Strain, Fatigue Limit, Cladding Wastage, Peak Cladding	4.2-1.1, 4.2-1.6, 4.2-1.8, 4.2-1.3, 4.2-1.4, 4.2-1.5, 4.2-2.3, 4.2-2.5, 4.2-3.1, 4.2-3.3, 4.2-3.4, 4.2-3.5

Category	High-importance Phenomena/Parameters	Overall Knowledge Level	Applicable Design Limit	Applicable RAC
			Thermal Creep Strain	
	Number of strain cycles on cladding	K	Fatigue Limit	4.2-1.2
	Magnitude of strain cycles	P	Fatigue Limit	4.2-1.2
	Residence Time	P	Cladding Wastage	4.2-1.3, 4.2-1.4, 4.2-1.5, 4.2-2.5
	Detailed pin level irradiation histories including power and cladding temperature	K	Peak Cladding Temperature, Peak Fuel Temperature	4.2-1.11, 4.2-1.12, 4.2-2.1, 4.2-3.2, 4.2-2.2,4.2-3.4
	Detailed coolant transient temperature and pin power histories	P	Peak Cladding Temperature, Peak Fuel Temperature	4.2-1.11, 4.2-1.12, 4.2-2.1, 4.2-3.2, 4.2-2.2,4.2-3.4
Absorber Pin Parameters/ Operating Conditions	Depletions	P	Total Peak Cladding Strain, Peak Cladding Temperature Limit, Peak Absorber Temperature Limit	4.2-1.1, 4.2-1.7, 4.2-1.8, 4.2-1.11, 4.2-4.1
	Absorber thermal conductivity	P	Peak Absorber Temperature Limit	4.2-1.11, 4.2-4.1
	DPA on clad	P	Fatigue Limit, Total Peak Cladding Strain	4.2-1.2, 4.2-4.1
	Cladding temperatures	P	Total Peak Cladding Strain, Cladding Wastage Limit,	4.2-1.1, 4.2-1.2, 4.2-1.3, 4.2-1.4, 4.2-1.5, 4.2-1.7, 4.2-1.8, 4.2-1.11, 4.2-4.1
	Coolant temperatures	P	Peak Cladding Temperature Limit, Peak Absorber Temperature Limit	4.2-1.11
	Number of strain cycles on cladding	P	Fatigue Limit	4.2-1.2
	Magnitude of strain cycles	P	Fatigue Limit	4.2-1.2
	Residence time	P	Cladding Wastage	4.2-1.3, 4.2-1.4, 4.2-1.5, 4.2-1.11
	Sodium velocity	P	Fatigue Limit	4.2-1.2
	Bundle design/compliance	P	Fatigue Limit	4.2-1.2
	Detailed pin level irradiation histories	K	Peak Cladding Temperature, Peak	4.2-4.1

Category	High-importance Phenomena/Parameters	Overall Knowledge Level	Applicable Design Limit	Applicable RAC
	including power and cladding temperature		Absorber Temperature	
	Detailed transient power and coolant temperature histories	P	Peak Cladding Temperature, Peak Absorber Temperature	4.2-4.1
	Weld susceptibility to irradiation (embrittlement and swelling)	U	Total Peak Cladding Strain,	4.2-1.1, 4.2-1.7, 4.2-1.8, 4.2-1.11
	Na-bonded versus He-bonded	P	Peak Absorber Temperature Limit	4.2-1.11
	B4C to cladding gap characteristics	K	Peak Absorber Temperature Limit	4.2-1.11

Table 6-4. Summary of Identified High-Importance Phenomena and Associated Design Limits and Applicable RAC for Fuel and Control Assemblies

Category	High-Importance Phenomena/Parameters	Overall Knowledge Level	Applicable Design Criteria	Applicable RAC
Fuel System Damage under Normal Operation and AOOs	Impact loads due to handling drop accidents	K	Stress, Strain, Loading Limit	4.2-1, 4.2-3, 4.2-4
	Withdrawal/insertion forces	K	Stress, Strain, Loading Limit	4.2-1, 4.2-3, 4.2-4, 4.2-4
	Pin bundle to duct interaction	P	Dimensional changes	4.2-1, 4.2-3,
Core Seismic Criteria under Operating Basis Earthquake	Fuel assembly and component residual horizontal deformations	P	Reactivity Insertion Limit - Post-OBE Operability	4.2-2, 4.2-3, 4.2-4
	Fuel assembly and component residual horizontal displacements	P	Refueling Force Limit – Post-OBE Operability	4.2-2, 4.2-3, 4.2-4
Core Seismic Criteria under Safe Shutdown Earthquake	Fuel assembly and component horizontal displacements	P	Reactivity Insertion Limit – Pre-SCRAM displacements	4.2-2, 4.2-3, 4.2-4
	Fuel assembly and component residual deformations	P	Core Coolability Limit – Core Coolability	4.2-2, 4.2-3, 4.2-4

As described in RAC 4.2-6, three methods are acceptable for demonstrating that the fuel system design bases are met: 1) historic operating experience (Section 6.2), 2) testing (Section 6.3), and 3) analytical predictions (Section 6.4). The following subsections describe the current plan for using each of these methods to address the various fuel system design bases. More details will be provided for how these respective methods address the Experimental Data Assessment Framework and Evaluation Model Framework Goals from NUREG-2246 in the subsequent sections, where applicable.

6.2 Historic Operating Experience

The key design parameters available to improve the performance of fuel pins are related to the material used for fuel (metal fuel composition) and cladding (316 SS, D9[1], HT9) and fuel pin dimensions (smear density, cladding thickness, pin diameter, fuel column and plenum lengths). This section describes the historical experience and motivation for key design decisions. The Natrium Reactor Type 1 fuel is similar to fuel pins used historically and will target operating conditions within the operating envelope of earlier reactors.

It was recognized early in the development of metallic fuel pins that the key phenomena contributing to steady-state fuel pin performance include the fuel swelling and axial growth, fission gas release, cladding irradiation creep and swelling, FCCI and constituent redistribution in the fuel due to fission product generation and thermal gradients. Transient performance is also dependent on the fuel melting temperature, the formation of low melting point fuel/clad eutectics and thermal creep and total strain in the cladding. This section will describe how these key phenomena drove fuel pin design to achieve higher-burnup and longer residence times, while also ensuring performance during operational transients.

Metal fuels have high thermal conductivity and excellent compatibility with sodium relative to oxide fuels. In addition, transient testing at the TREAT reactor demonstrated that despite having a lower melting point than oxide fuels, metal fuels performed exceptionally under fast transient conditions. For example, metal fuel failed at five times the nominal peak power during TREAT transient testing[2] [5]. PIE studies on early [43], low-burnup fuel pins indicated that adding small amounts of alloying elements, such as zirconium, to metallic fuel increased dimensional stability but could not stop fission gas induced swelling of the fuel. Zirconium became the alloying element of choice because it also increased the fuel solidus temperature and increased the fuel/cladding eutectic formation temperature above that of the binary uranium/iron alloy (725°C (1337°F)) [44] for fuel with less than 10 wt% plutonium [45]. Natrium Reactor Type 1 fuel has uranium-10 wt % zirconium fuel, which was used in early reactors in the U.S.A.

Fuel swelling limited the burnup capability of early metal fuel pin designs, and it was recognized that the swelling was caused by the accumulation of fission products [46]. It was also noted that fission gas is essentially insoluble in metal fuels and that at pore volume fractions over 30% the pores become linked effectively releasing the fission gas [47]. These observations led to a second generation of fuel pins that had 75% smear density and greatly increased plenum lengths (up to 1.45 times the fuel volume). These pins achieved up to ~10% burnup. Fuel pins with 75% smear density also included a sodium bond layer to conduct heat between the fuel and the clad at the beginning of life before the fuel contacts the clad. The fuel grows in the axial direction ~2-10% before the fuel and cladding are in direct contact. After contact between the fuel and cladding, axial growth changes are minimal. Natrium Reactor Type 1 fuel incorporates the smear density of 75% and the larger plenum volume that was developed in these second-generation pins.

The third generation of fuel pins, designed and fabricated in the late 1980s, incorporated ferritic-martensitic steels such as HT9. These improved cladding alloys exhibited low swelling from internal void formation. In addition, they show superior resistance to irradiation creep. The pin dimensions

[1] D9 is a nuclear grade austenitic stainless steel based on AISI type 316-SS that has improved void-swelling resistance.
[2] Metal fuel experiments failed during transient testing at more than 200 kW/m, which is approximately [[ ]]$^{(a)(4)(ECI)}$ times the nominal linear power of Type 1 fuel in the Natrium Reactor.

also evolved to include a 33% increase in cladding thickness, a larger diameter, and longer pin length. The structural load on the cladding is due largely to fission products. Fission gases drive the initially high rate of fuel swelling. The porosity eventually becomes interconnected allowing the gases to access the plenum volume. Eventually, the porosity beings to fill with solid fission products and the fission gas is concentrated in the interconnected porosity and plenum region increasing the load on the cladding at high burnups. Natrium Reactor Type 1 fuel pins have HT9 cladding.

FCCI is a phenomenon that may limit the lifetime of metal fuel pins, particularly if the operating temperatures are relatively high. It is caused by the diffusion of lanthanide fission products into the clad which can lead to embrittlement. The embrittlement is so severe that the layer of impacted cladding is considered unable to support any loads. During structural analysis, the cladding thickness is reduced by the width of the FCCI layer (thinning of the clad is assumed to occur). Therefore, FCCI impacts fuel performance during normal operations and transients. The Natrium Reactor Type 1 fuel pins will be used within the burnup and operational limits of the historical database.

Another diffusional process that impacts material properties and performance in fuel pins is constituent redistribution in the fuel. Depending on operating conditions, zirconium may migrate up and down the thermal gradient in alloy fuel, which may lead to a higher concentration of zirconium at the center of the fuel pin and at the inner surface of the cladding, with corresponding depletion in the intermediate region. This can impact the local solidus temperature due to the dependence on zirconium concentration. Peak fuel temperatures occur at the center of the fuel where zirconium concentrations are seen to rise due to constituent redistribution, slightly increasing solidus temperatures relative to beginning of life conditions. Natrium Type 1 fuel is expected to have less pronounced constituent redistribution than was typical for the EBR-II and FFTF metallic fuels due to a lower targeted pin power which leads to smaller temperature gradients.

Metallic fuel pins were used in fuel assemblies in the EBR-I and EBR-II reactors. A metallic fuel qualification test program was in progress for the FFTF, known as the MFF[1] series of fuel assemblies, when it was shut down in 1994. The FFTF fuel designs were more relevant to commercial designs because of their larger diameter, longer length fuel pins and fuel assemblies with 169 fuel pins compared to 61 pins in EBR-II. The MFF pins had 75% smear density uranium-10 wt % zirconium and HT9 cladding. Burnups up to ~15 %FIMA were tested with a range of temperatures from nominal up to 2-sigma hot channel factor[2]. The total fission gas release was consistent with that measured for shorter pins. HT9 cladding had 6-10 times less diametral strain relative to D9, which was attributed to the lower swelling of HT9 [48]. Results from PIE show that the FCCI time-temperature data and nonuniform circumferential depth for the MFF pins was also consistent with the shorter EBR-II pins; however, in some cases the depth of the FCCI layer did not reach a maximum at the top of the fuel column [48].

The metal fuel irradiation operational experience consists of ~130,000 metal fuel pins with most of the burnups less than 10 %FIMA but with some as high as 20 %FIMA burnup. The pins can be divided into approximately four categories:

[1] A recent reference identified MFF as Mechanistic Fuel Failure [48] but no official definition was found in the original literature from FFTF.

[2] The hot channel factor comes from the hot channel model where the hot channel is affected by the simultaneous occurrence of all uncertainties [84].

- 90,000 pins with 85% smear density or higher and 2.6 %FIMA burnup
- 30,000 pins with 75% smear density and 8 %FIMA burnup
- 13,000 pins with 75% smear density and larger diameter with 10 %FIMA burnup
- 1,050 pins developed for the FFTF/MFF, metallic fuel qualification tests with 75% smear density, larger diameter and HT9 cladding with varying burnup.

The EBR-II fuel performance was exceptional. Out of 30,000 pins of an early design (90% smear density, uranium-2 wt.% zirconium fuel, SS347 cladding) there were only 13 failures related to a restrainer on the pin that was removed from the design eliminating this failure mode. 13,600 pins with alloy fuel were used with only 22 natural breached pins reported (19 were in high temperature tests run to failure with 16 of the reported failures at welds). Seven tests to run metallic fuel pins beyond cladding breach were also performed in EBR-II. The tests, which ran between 34 and 233 days post cladding breach, indicated that almost all the fission gas in the plenum and interconnected porosity is released into the primary system during the breach. The post-breach fuel behavior was benign and breached pins showed little difference from intact pins regardless of the post-breach operating time.

EBR-II transient tests were also performed which demonstrated operational reliability including 56 tests with a low ramp rate (1.6 % power increase per second) and 13 with a high ramp rate (6.3 % power increase per second) [49]. More severe transient tests demonstrated that metal fueled cores can safely operate with loss-of-flow-without-SCRAM events and loss-of-heat-sink-without-SCRAM events with no core damage. These tests were performed with fully instrumented and calibrated in-core fuel assemblies and showed that the fuel pins successfully operated for 42 minutes with cladding temperatures as high as 800°C.

Transient tests on fuel pins were also performed at the TREAT reactor at Idaho National Laboratory. TREAT is capable of performing tests on individual fuel pins that are extremely severe, such as melting the fuel or breaching the cladding, to understand failure mechanisms more fully in fuel pins under severe conditions. In general, fuel pins will fail by one of the following mechanisms based on burnup.

Fuel Clad Thermal Creep Strain Failure: Failure occurs due to localization of thermal creep strain (creep rupture) driven by high pressure fission gas in the plenum and open porosity in the fuel due to the high temperatures during the transient. This failure mechanism is seen where high cladding temperatures are sustained and fuel burnup is high enough that the stress imposed by the fission gas leads to high diametral strains in the cladding.

Fuel Cladding Total Wastage Failure: Failure occurs due to accelerated degradation of the cladding from the formation of eutectic phases. These reactions can occur extremely quickly (on the order of minutes) and occur in lower burnup pins that do not have a large inventory of fission gas.

Fuel Melting: Occurs when fuel temperatures exceed the melting temperature of the fuel. Fuel melting also contributes to eutectic formation by supplying free uranium for reactions occurring at the fuel-cladding interface. This failure mode may occur at a range of burnup levels; however, requires very significant overpower conditions.

A total of 15 transient overpower tests of metallic fuel were performed at TREAT and show similar trends. The differences between alloys and samples can be largely explained by retained fission gas and variation in melting temperature between the fuel alloys. In general, failures occurred at the top of the fuel column where the inner cladding temperature tends to be high relative to the bottom of the fuel column. Two trends were noted that may mitigate the impact of fuel pin failures in a reactor. The first is that the fuel expands axially due to thermal expansion and from the expansion of fission gas in porosity in the fuel column. Expansion of the fuel in the axial direction generally reduces the reactivity of the core. The second is that fuel dispersal occurred quickly after a cladding breach, also removing reactivity from the core.

Although there is no direct commercial reactor operating experience with Type 1 fuel, the fuel design and operational targets are intentionally modeled after metallic fuel designs that successfully operated in EBR-II and FFTF, providing confidence in the overall reliability of the fuel design. Table 6-5 provides a comparison of the fuel pin dimensions and other design parameters for Type 1 Fuel, FFTF, and EBR-II fuel pins. Table 6-6 gives a similar comparison for Natrium Reactor absorber pins and FFTF absorber pins [50]. Table 6-7 provides a comparison between Type 1 fuel assemblies and other SFR assemblies. Using the recent Natrium fuel performance assessment to support conceptual design [51], comparisons of anticipated Type 1 fuel pin conditions and corresponding conditions from EBR-II and FFTF are summarized in Table 6-8. Peak burnup and linear heat generation rate are within the range of the tested parameters in EBR-II and FFTF, [[

]](a)(4)(ECI). Additionally, a fuel surveillance program is planned to monitor performance at interim burnup steps (covered by this historic database) to verify performance is consistent with predictions (see Section 9).

Table 6-5. Summary of Fuel Pin Parameters Including Comparison to FFTF/MFF[1] and EBR-II

Parameter	Unit	Type 1 Fuel	FFTF/ MFF [54]	EBR-II		
				MkIII Driver Fuel [16]	**MkIV Driver Fuel** [16]	**X430** [55]
Cladding Outer Diameter	in.	[[ ]](a)(4)(ECI)	0.270	0.23	0.23	0.29
	mm	[[ ]](a)(4)(ECI)	6.86	5.84	5.84	7.37
Cladding Thickness	in.	[[ ]](a)(4)(ECI)	0.022	0.015	0.018	0.016
	mm	[[ ]](a)(4)(ECI)	0.56	0.38	0.46	0.41
Cladding Thickness/ Diameter Ratio		[[ ]](a)(4)(ECI)	0.081	0.065	0.078	0.055
Active Fuel Column Length	in.	[[ ]](a)(4)(ECI)	36.0	13.5	13.5	13.5
	m	[[ ]](a)(4)(ECI)	0.914	0.343	0.343	0.343
Plenum Length	in.	[[ ]](a)(4)(ECI)	51.2	14.7	14.7	14.5*
	m	[[ ]](a)(4)(ECI)	1.3	0.373	0.373	0.368
Fuel Type	-	U-10Zr [30]	U-10Zr	U-10Zr	U-10Zr	U-10Zr, U-Pu-10Zr
Weight Fraction Zr	-	0.1	0.1	0.1	0.1	0.1
Nominal Fuel Smear Density Fraction	-	0.75	0.75	0.75	0.75	0.75
Bond	-	Sodium [30]	Sodium	Sodium	Sodium	Sodium
Cladding Material	-	HT9 [56]	HT9	D9	HT9	HT9

*Derived Value

Table 6-6. Summary of Absorber Pin Parameters Including Comparison to FFTF and JOYO

Parameter	Unit	Natrium [31]	FFTF (Series 1) [57]	FFTF (Series 2) [57]	FFTF (Series 3) [57]	JOYO Mk-II [58]
Control Rod Absorber Material	-	B_4C (19.9 at% $^{10}B/B$)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	B_4C (90 at% $^{10}B/B$)
Smear Density	%	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	93
Cladding Tube Material	-	HT9	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	PNC 316
Fill Gas	-	He	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	He
Cladding Outer Diameter	in.	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	0.717
	mm	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	18.2
Cladding Thickness	in.	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	0.032
	mm	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	0.8
Cladding Thickness/ Diameter Ratio	-	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	0.045

[1] FFTF was fueled with oxide fuel, but eight fuel assemblies using metallic fuel were fabricated to support eventual conversion to metallic fuel. These assemblies were labeled MFF, but no official definition for the label/acronym has been identified.

Parameter	Unit	**Natrium** [31]	**FFTF (Series 1)** [57]	**FFTF (Series 2)** [57]	**FFTF (Series 3)** [57]	**JOYO Mk-II** [58]
B_4C Column Length	in.	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	25.6
	mm	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	650
B_4C Pellet Diameter	in.	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	0.630
	mm	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	16
Number of Pins/Assembly	-	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	7
Total Plenum length	in.	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	17.7
	mm	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	449
Wire wrap diameter	in.	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	0.098
	mm	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	2.5
Overall Control Pin Length	in.	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	50.12
	m	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	1.273
Design Lifetime	EFPD	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	-

Table 6-7. Type 1 Fuel Assembly Design Parameters

Assembly Design Parameters	**Unit**	**Type 1 Fuel**	**FFTF III.b** [59] [17]	**PRISM** [60]	**VTR** [61]
Duct Material	-	HT9 [62]	HT9	HT9	HT9
Duct Thickness	in.	[[ ]](a)(4)(ECI)	0.118	0.140	0.118
	mm	[[ ]](a)(4)(ECI)	3.00	3.556	3.00
Duct Flat to Flat OD	in.	[[ ]](a)(4)(ECI)	4.59	6.106	4.606
	mm	[[ ]](a)(4)(ECI)	116.5	155.1	117
Number of Pins/Bundle	-	[[ ]](a)(4)(ECI)	169	271	217
Pin Length	in.	[[ ]](a)(4)(ECI)	93.4	158.0	64.96
	m	[[ ]](a)(4)(ECI)	2.371	4.013	1.650
Inlet Nozzle / Nosepiece	in.	[[ ]](a)(4)(ECI)	33.7	13.0	12.99
	mm	[[ ]](a)(4)(ECI)	856.0	330.2	329.9
Assembly Total Length	in.	[[ ]](a)(4)(ECI)	144	186	152
	m	[[ ]](a)(4)(ECI)	3.658	4.724	3.861
Fuel Assembly Weight	lbs.	[[ ]](a)(4)(ECI)	395.6	[1]	[1]
	kg	[[ ]](a)(4)(ECI)	179.4	[1]	[1]

[1] Information is currently unavailable

Assembly Design Parameters	**Unit**	**Type 1 Fuel**	**FFTF III.b** [59] [17]	**PRISM** [60]	**VTR** [61]
Number of Assemblies/ Core	-	[[ ]](a)(4)(ECI)	74	99	110
Fuel Column Length	in.	[[ ]](a)(4)(ECI)	36.0	47.0	31.5
	mm	[[ ]](a)(4)(ECI)	914.4	1193.8	800.0
Wire Wrap Diameter	in.	[[ ]](a)(4)(ECI)	0.054	0.056	0.044
	mm	[[ ]](a)(4)(ECI)	1.36	1.42	1.11
Wire Wrap Axial Pitch	in.	[[ ]](a)(4)(ECI)	9.20	12.00	10.51
	mm	[[ ]](a)(4)(ECI)	233.7	304.8	267
Fuel Pin Pitch/Diameter (p/d) Ratio	-	[[ ]](a)(4)(ECI)	1.20	1.199	1.18
Assembly Pitch	in.	[[ ]](a)(4)(ECI)	-	6.283	4.724
	mm	[[ ]](a)(4)(ECI)	-	159.6	120.0

Table 6-8. Comparison of Fuel System Operational Parameters

Parameter	**Type 1 Fuel Bounding Operating Values** [51]	**FFTF (MFF-2)** [63]	**FFTF (MFF-3)** [64]	**EBR-II (X447)** [52]	**EBR-II (X425)** [52]
Enrichment, % U-235	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)
Peak Burnup, %FIMA	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)
Peak, DPA	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)
Residence, EFPD	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)
Peak Linear Heat Rate, kW/m	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)
Peak Inner Cladding Temperature, °C	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)
Peak Inner Cladding Temperature, °F	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)
Peak Fast Fluence, (E>0.1 MeV) n/cm^{2}s	[[ ]](a)(4)(ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)	[[ ]](ECI)

[1] This value represents the 2σ hot channel factor (HCF) temperature

The axial distribution for parameters such as linear heat generation rate, burnup, and DPA generally follow a Gaussian shape, shifted slightly towards the bottom of the core due to control rod insertion from the top of the core, as seen in Figure 6-2 for a nominal pin. Over a given cycle, the power shifts slightly towards the top of the core as control rods are withdrawn due to a change in reactivity from the depletion over the cycle. The axial burnup of Type 1 fuel is more peaked in comparison to EBR-II and FFTF, although more similar to FFTF (Figure 6-1) [65]. This has the attractive structural performance feature of reducing burnup at the highest temperature conditions, which reduces the load on the cladding where FCCI and temperature is the greatest. The axial temperature distribution at beginning of life is shown for different temperature conditions in Figure 6-3.

(a)(4)(ECI)

Figure 6-1. Burnup Distribution Comparison between MFF, EBR-II, and Natrium Fuel Pins[1]

(a)(4)(ECI)

Figure 6-2. Fuel Linear Heat Generation Rate Distribution (left) and Burnup Distribution (right) for a Nominal Pin in the Inner Core Region

[1] Burnup distributions taken from representative high-burnup fuel pins. EBR-II calculated values come from ANL's FIPD database [52], [[ ]](a)(4).

[[ ]](a)(4)(ECI)

Figure 6-3. Cladding Surface Temperature Distribution at Beginning of Life

All applicable fuel tests from EBR-II and FFTF have been reviewed and a detailed plan has been developed to identify and prioritize the fuel pins/assemblies of interest [66]. From EBR-II, [[ ]](a)(4) have been selected for data qualification. From FFTF, [[ ]](a)(4) from the MFF series tests are also included. These pins span a wide range of design parameters and operating conditions that generally bound the operating conditions targeted for Type 1 fuel. For each subassembly, quantities of measured data from profilometry, gamma scans, neutron radiography (NRAD), and gas release measurements have been determined. Profilometry data will primarily be used to validate predictions of cladding strain, therefore a focus is placed on fuel pins with HT9 cladding at a wide range of operating conditions. NRAD and gas release measurements focus on U-10Zr fuel pins. NRAD data will be used for validation of radial and axial fuel swelling. Gas release measurements will be used for validation of predicted gas release fractions and plenum pressurization. Cladding wastage measurements from fuel pin metallography will be used to validate wastage predictions. Fuel pin data from transient testing (e.g., TREAT, furnace tests) has similarly been identified for benchmarking predictions of fuel pin failure, fuel radial melt locations, and fuel axial expansion due to fuel melting. Benchmark comparisons to material test data (e.g., tube burst, creep) is similarly planned. Validation comparisons to absorber pin data such as plenum pressure, gas release fraction, B_4C swelling, ACCI, and B_4C temperature will also be performed. A summary of the targeted assemblies, relevant design parameters, operating conditions, and quantities of measured post irradiation exam data for fuel pins and absorber pins is provided in Table 6-9 and Table 6-10, respectively.

Table 6-9. Relevant Historic Fuel Assemblies to Support Validation Activities

	Operating Conditions[1,2]						Measured PIE Data Quantities						Design Characteristics										
Subassembly	Peak Burnup [%]	Peak Linear Power [kW/m]	Peak Inner Clad Temperature [°C]	Peak Inner Clad Temperature [°F]	Peak DPA [-]	Peak Pin Residence [EFPD]	Contact Profilometry	Laser Profilometry	Gamma Scan	NRAD	Fission Gas Release	Fission Gas Chemistry	Cladding Material	Zr %	Pu %	Zr-Sheathed Fuel	Fuel Outer Diameter [in]	Fuel Length [in]	Plenum to Fuel Volume Ratio [-]	Cladding Outer Diameter [in]	Cladding Thickness [in]	Nominal Smear Density [-]	Qualification Priority
X419																							
X419A																							
X419B																							
X420																							
X420A																							
X420B																							
X421																							
X421A																							
X423																							
X423A																							
X423B																							
X423C																							
X425																							
X425A																							
X425B																							
X425C																							
X429																							
X429A																							
X429B																							
X430																							
X430A																							
X430B																							
X431																							
X431A																							
X432																							
X432A																							
X441																							
X441A																							
X447																							

(ECI)

[1] Operating conditions for EBR-II fuel pins come from ANL's FIPD database [52]. [[ ]](a)(4).
[2] Operating conditions for MFF fuel pins were analyzed by [[ ]](a)(4) [64] [63]. [[ ]](a)(4).

	Operating Conditions[1,2]						Measured PIE Data Quantities						Design Characteristics											
Subassembly	Peak Burnup [%]	Peak Linear Power [kW/m]	Peak Inner Clad Temperature [°C]	Peak Inner Clad Temperature [°F]	Peak DPA [-]	Peak Pin Residence [EFPD]	Contact Profilometry	Laser Profilometry	Gamma Scan	NRAD	Fission Gas Release	Fission Gas Chemistry	Cladding Material	Zr %	Pu %	Zr-Sheathed Fuel	Fuel Outer Diameter [in]	Fuel Length [in]	Plenum to Fuel Volume Ratio [-]	Cladding Outer Diameter [in]	Cladding Thickness [in]	Nominal Smear Density [-]	Qualification Priority
X447A																							
X492																							
X492A																							
X492B																							
X496																							
MFF2																							
MFF3																							
MFF5																							
MFF6																							
Max																							
Total																							

(ECI)

Table 6-10. Relevant Historic Absorber/Control Pin Test Assemblies to Support Validation Activities

Experiment Details			Operating Conditions					Measured PIE Data									Design Characteristics							
Experiment/Assembly	Reactor	# Pins	Peak Depletion [10^{20} cap/cc]	Time (EFPD)	Power (MWD)	Peak Pellet Temperature [°C]	Fluence (10^{22} n/cm^2)	Irradiation History	Profilometry	Swelling	Gas Release	Cracking	Max Plenum Pressure (psi)	Radiographs	TEM/SEM	ACCI	Cladding Material	Cladding Outer Diameter [mm]	Cladding Outer Diameter [in]	Cladding Thickness [mm]	Cladding Thickness [in]	Theoretical Density [%]	B-10 Enrichment [%]	Smear Density
CR-1 (CRA-528)	FFTF																							
CR-2 (CRA-541)	FFTF																							
CR-3 (CRA-537)	FFTF																							
BICM-1 (YY02)	EBR-II																							
BICM-2 (YY06)	EBR-II																							
WDC-1-1	ETR																							
BOPT-1 (X-248/248A/249)	EBR-II																							
BOPT-2 (large diameter)	EBR-II																							
BOPT-2 (small diameter)	EBR-II																							
BV-2A (X-256)	EBR-II																							
BV-2B (X-265)	EBR-II																							
BV-2C (X-257)	EBR-II																							
He-bonded	JOYO Mk-II																							
He-bonded with shroud	JOYO Mk-II																							
Na-bonded with shroud	JOYO Mk-III																							
ADVAB-1	FFTF																							
ADVAB-2	FFTF																							
Vented Pins	EBR-II																							
CR-7	FFTF																							
CR-8	FFTF																							

(ECI)

In addition to the reports summarizing data from specific fuel assemblies, reports have also been issued summarizing key fuel performance phenomena. Specifically, there are dedicated reports on [[

]](a)(4). Note that these reports cover all the high-importance phenomena summarized in Table 6-3 except for [[ ]](a)(4). There is limited [[conductivity]](a)(4). To address this lack of historical [[ ]](a)(4) additional testing is planned [[ ]](a)(4). See Section 6.2 for more details.

As shown in Table 6-5 and Table 6-8,the [[ ]](a)(4) assemblies [[ ]](a)(4) most closely match the Type 1 fuel design and targeted operational conditions; [[

]](a)(4). The measured cladding strains on these pins will be the primary independent validation basis for TerraPower's fuel performance tools, with a subset of the fuel pins excluded from model development/calibration efforts to serve as a blind comparison. Specific references for the data collected thus far will be included in the applicable areas in subsequent sections.

6.2.1 Quality of Historic Data

The ability to leverage historic data greatly reduces the amount of testing required; however, an assessment is required to verify that the historic data have been accurately measured (NUREG-2246 Experimental Data (ED) Assessment Framework Goal ED G3). NUREG-2246 identifies three sub-goals to demonstrate the accurate measurement of experimental data: ED G3.1) the test facility has an appropriate quality assurance program, ED G3.2) experimental data are collected using established measurement techniques, and ED G3.3) experimental data account for sources of experimental uncertainty.

Both EBR-II and FFTF were DOE test reactors with rigorous quality programs; however, because the data was not collected under TerraPower's approved quality assurance program [75], additional effort is required to qualify the existing data for use. Specifically, four methods are approved for qualifying existing data: 1) Demonstrate Quality Assurance Program Equivalency, 2) Data Corroboration, 3) Confirmatory Testing, or 4) Peer Review [76].

To support this effort, data qualification plans will be developed for each set of historic data requiring qualification. Specifically, for EBR-II and FFTF experience, a data qualification plan has been developed to evaluate and qualify the historical or pre-existing fuel and absorber pin data intended for use to support model development and validation [66]. As part of this plan Argonne National Laboratory (ANL) will take the lead for qualifying the relevant EBR-II fuel pin steady-state and transient data relying on the process that was previously reviewed and approved by the NRC [77] [78].

6.3 Testing

As shown in Section 6.2, the operating experience of metallic fuels in EBR-II and FFTF largely covers the targeted conditions for Type 1 fuel, and the intention is to rely heavily on this basis for the safety case and validation of methods for Type 1 fuel pins; however, a subset of tests have been or will be performed to supplement the historic operating experience. These tests primarily fall within five categories: 1) verification of consistent (or improved) performance between newly manufactured materials/ and historic materials, 2) testing to cover extrapolations of conditions beyond the historic database (i.e., long-duration creep testing), 3) testing to reduce uncertainties or improve fundamental understanding, 4) testing requiring prototypic bundle or fuel pin geometries, and 5) testing to address any gaps in the historic operational experience [[ ]]$^{(a)(4)}$. These tests include separate effect and integral effect tests. Summaries of the testing that has been performed to support Type 1 fuel qualification are given in Table 6-11 through Table 6-13 organized according to RAC. Future tests that are envisioned to address Fuel Damage, Fuel Failure, and Fuel Coolability Criteria are given in Table 6-14 through Table 6-16. A summary of high-priority testing activities specifically related to the high-importance phenomena is provided in Table 6-17. Note that some of these identified tests may be eliminated pending additional analysis or retrieval of additional historic data.

Table 6-11. Design Basis Criteria and Supporting Information to Prevent Fuel Pin Damage

Specific RAC	Applicable Design Basis Criteria	Applicable Pin	Available Supporting Data
4.2-1.1	Total Diametral Strain of Cladding	Fuel/Absorber	[[ ]]$^{(a)(4)}$
		Fuel/Absorber	[[ ]]$^{(a)(4)}$
		Fuel/Absorber	[[ ]]$^{(a)(4)}$
		Fuel/Absorber	[[ ]]$^{(a)(4)}$
		Absorber	[[ ]]$^{(a)(4)}$
4.2-1.2	Design Fatigue Lifetime	Fuel/Absorber	[[ ]]$^{(a)(4)}$
4.2-1.3	Cladding Wastage (Fretting)[1]	Fuel/Absorber	[[ ]]$^{(a)(4)}$
		Fuel/Absorber	[[ ]]$^{(a)(4)}$
4.2-1.4	Cladding Wastage (Na Corrosion)[1]	Fuel/Absorber	[[ ]]$^{(a)(4)}$
		Fuel/Absorber	[[ ]]$^{(a)(4)}$
4.2-1.5	Cladding Wastage (FCCI) or (ACCI)[1]	Fuel	[[ ]]$^{(a)(4)}$
		Fuel	[[ ]]$^{(a)(4)}$

[1] A single limit is set for wastage; however, all contributions (fretting, Na corrosion, FCCI/ACCI) are assessed in a conservative manner and combined to verify the total wastage limit is not exceeded.

Specific RAC	Applicable Design Basis Criteria	Applicable Pin	Available Supporting Data
		Absorber	[[ ]](a)(4)
4.2-1.6	Total Diametral Strain of Cladding	Fuel	See activities from 4.2-1.1
4.2-1.7	Total Diametral Strain of Cladding	Absorber	See activities from 4.2-1.1
4.2-1.8	Total Diametral Strain of Cladding requires internal fuel pin pressure to be assessed	Fuel	[[ ]](a)(4)
		Fuel/Absorber	[[ ]](a)(4)
		Fuel/Absorber	[[ ]](a)(4)
		Absorber	[[ ]](a)(4)
4.2-1.11[1]	Peak Cladding Temperature	Absorber	[[ ]](a)(4)
	Peak Absorber Temperature	Absorber	[[ ]](a)(4)
	Cladding Wastage	Absorber	See activities from 4.2-1.3, 4.1-1.4, 4.2-1.5
	Total Diametral Cladding Strain	Absorber	See activities from 4.2-1.1
4.2-1.12	Peak Cladding Temperature	Fuel	[[ ]](a)(4)
		Fuel	[[ ]](a)(4)
	Peak Cladding/Absorber Temperature	Absorber	[[ ]](a)(4)

Table 6-12. Design Basis Criteria and Supporting Information to Predict Fuel Failure

Specific RAC	Applicable Design Basis Criteria	Available Supporting Data
4.2-2.1	Peak Cladding Temperature	[[ ]](a)(4)
4.2-2.2	Peak Fuel Temperature	[[ ]](a)(4)
4.2-2.3	Thermal Creep Strain of Cladding	[[ ]](a)(4)
		[[ ]](a)(4)
		[[ ]](a)(4)
		[[ ]](a)(4)

[1] Note RAC 4.2-1.9 and 4.2-1.10 are not applicable to fuel or absorber pins

Specific RAC	Applicable Design Basis Criteria	Available Supporting Data
4.2-2.5	Cladding Wastage (FCCI)[1]	See activities from 4.2-1.5
	Cladding Wastage (Eutectic)[1]	See activities from 4.2-2.1
	Cladding Wastage (Na Corrosion)[1]	See activities from 4.2-1.4
	Cladding Wastage (Fretting)[1]	See activities from 4.2-1.3

Table 6-13. Design Basis Criteria and Supporting Information to Ensure Fuel Pin Coolability and Absorber Pin Insertability

Specific RAC	Applicable Design Basis Criteria	Applicable Pin	Available Supporting Data
4.2-3.1	Total Diametral Strain of Cladding	Fuel	See activities from 4.2-2.3
4.2-3.2	Peak Cladding Temperature	Fuel	See activities from 4.2-2.1
4.2-3.3	Total Diametral Strain of Cladding	Fuel	See activities from 4.2-2.3
4.2-3.4	Peak Fuel Temperature	Fuel	See activities from 4.2-2.2
4.2-4.1	Total Diametral Strain	Absorber	See activities from 4.2-1.11
	Peak Absorber Temperature	Absorber	
	Peak Cladding Temperature	Absorber	

[1] A single limit is set for wastage; however, all contributions (fretting, eutectic interactions, Na corrosion, FCCI) are assessed in a conservative manner and combined to verify the total wastage limit is not exceeded.

Table 6-14. Summary of Future Testing Activities to Validate Fuel Damage Limits

RAC	Design Basis Criteria	Identified Activity[1]	Main Objectives	Primary Factors of Concern	References
4.2-1.1	Total Diametral Clad Strain	[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	[107] [80] [79] [82] [81] [83]
		[[ ]](a)(4)	[[ ]](a)(4)		[70] [98]
		[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	[84] [86]
		[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	[108]
		[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	[109] [110]
4.2-1.2	Maximum allowable fuel pin fatigue cycles	[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	

[1] Note that some of these identified tests may be eliminated pending additional analysis or retrieval of additional historic data.

RAC	Design Basis Criteria	Identified Activity[1]	Main Objectives	Primary Factors of Concern	References
4.2-1.3	Cladding Wastage – Fretting	[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$		
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	[111] [112]
4.2-1.4	Cladding Wastage – Na Corrosion	[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	[74]
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$		
4.2-1.5	Cladding Wastage – FCCI	[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	[48] [74] [113]
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$		[114]
	Cladding Wastage – ACCI	[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[	[115]

RAC	Design Basis Criteria	Identified Activity[1]	Main Objectives	Primary Factors of Concern	References
				•]](a)(4)	
4.2-1.6	Total Diametral Cladding Strain	[[ ]](a)(4)			
4.2-1.8	Total cladding strain and thermal creep strain limits require internal fuel pin pressure to be assessed	[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	[74]
		[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	[114]

Table 6-15. Summary of Future Testing Activities to Predict Fuel Failure

RAC	Design Basis Criteria	Identified Activity[1]	Main Objectives	Primary Factors of Concern	References
4.2-2.1	Peak cladding temperature limit	[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	[116]
		[[ ]](a)(4)	[[ ]](a)(4)		
		[[ ]](a)(4)	[[	• [[	

[1] Note that some of these identified tests may be eliminated pending additional analysis or retrieval of additional historic data.

RAC	Design Basis Criteria	Identified Activity[1]	Main Objectives	Primary Factors of Concern	References
			]]$^{(a)(4)}$	]]$^{(a)(4)}$	
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	
4.2-2.2	Peak fuel temperature limit	[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[ ]]$^{(a)(4)}$	[117]
		[[ ]]$^{(a)(4)}$	[[ ]]$^{(a)(4)}$	• [[	

RAC	Design Basis Criteria	Identified Activity[1]	Main Objectives	Primary Factors of Concern	References
				]](a)(4)	
4.2-2.3	Cladding strain-thermal creep	[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	
		[[ ]](a)(4)	[[ ]](a)(4)		
		[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	
		[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)	
4.2-2.5	Cladding Wastage – FCCI	See activities from 4.2-1.5			
	Cladding Wastage – Eutectic	See activities from 4.2-2.1			
	Cladding Wastage –Na Corrosion	See activities from 4.2-1.4			
	Cladding Wastage – Fretting	See activities from 4.2-1.1			

RAC	Design Basis Criteria	Identified Activity[1]	Main Objectives	Primary Factors of Concern	References
	Cladding Wastage – ACCI	See activities from 4.2-1.5			

Table 6-16. Summary of Future Testing Activities to Ensure Fuel Coolability is Maintained

RAC	Design Basis Criteria	Identified Activity[1]	Main Objectives	Primary Factors of Concern
4.2-3.1	Total Diametral Cladding Strain-Coolability	[[ ]](a)(4)	[[ ]](a)(4)	• [[ ]](a)(4)
4.2-3.2	Peak Cladding Temperature	See activities from 4.2-2.1		
4.2-3.3	Total Diametral Cladding Strain-Coolability	See activities from 4.2-3.1		
4.2-3.4	Peak Fuel Temperature	See activities from 4.2-2.2		

[1] Note that some of these identified tests may be eliminated pending additional analysis or retrieval of additional historic data.

Table 6-17 Summary of Tests to Address High-Importance Fuel and Absorber Pin Phenomena

High-Importance Phenomena	Applicable Design Limit	Overview of Testing[1]
Fission gas release	Total Peak Cladding Strain, Peak Cladding Thermal Creep	• [[ ]](a)(4)
HT9 mechanical response as a function of temperature, stress, irradiation, and time	Total Peak Cladding Strain, Fatigue Limit, Peak Cladding Thermal Creep Strain	• [[ ]](a)(4)
FCCI/ACC	Cladding Wastage, Peak Cladding Thermal Creep Strain	• [[ ]](a)(4)
Fuel Thermal Conductivity	Peak Fuel Temperature	• [[ ]](a)(4)

6.3.1 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 will be developed to validate the numerical models of the fuel

[1] Note that some of these identified tests may be eliminated pending additional analysis or retrieval of additional historic data.
[2] [[ ]](a)(4).

assembly, which can be categorized into the following areas: 1) sub-component tests (SC), 2) single assembly tests (SA), 3) multiple assembly tests (MA), 4) control assembly tests (CA), and 5) design proof of concept tests (DPC). Test categories 1-4 are described in section 6.3.1.1 through 6.3.1.3 and Table 6-19, and DPC tests (5) are described in section 6.3.1.4. The envisioned tests for fuel and control assemblies are summarized below in these respective categories.

6.3.1.1 Sub-Component Tests

6.3.1.1.1 Bundle-Duct Interaction Tests

This series of tests will examine mechanical characteristics of the pin bundle including compression stiffness and pin redistribution under various loads. Data gathered from these tests will provide initial validation data in the elastic range and will be used for benchmarking of the OXBOW.BDI code.

6.3.1.1.2 Pin Bundle Bending Tests

The purpose of this test is to gather initial pin bundle stiffness information and validate assumptions which are made in some core mechanical models. Additionally, bundle stiffness values gathered from this test can be useful in developing preliminary pin bundle models for other analyses.

6.3.1.1.3 Core Mechanical Duct Static Crush Tests

The purpose of this test is to obtain the static compressive strength of the duct and load pads for use as a limit in core restraint system analysis. This would prevent damage to core assembly pin bundles.

6.3.1.1.4 Duct Dynamic Crush Tests

The purpose of this test is to obtain the dynamic crush strength of the duct and load pads for use as a limit in seismic analyses. This would prevent damage to core assembly pin bundles.

6.3.1.1.5 Nozzle/Receptacle Interaction Tests

The purpose of this test is to investigate how an assembly nozzle interacts with its receptacle due to applied forces and moments. The lateral and rotational stiffness of this interaction dictates the displacements of core assemblies.

6.3.1.2 Single Assembly Tests

6.3.1.2.1 Single Assembly Static Load Deflection Test

The purpose of this series of tests is to investigate and characterize the mechanical behavior of SFR core assemblies. Tests will focus on mechanical behavior (bending stiffness, range of motion, etc.) of single fuel and control assemblies in the elastic range as well as thermal effects. This will include both nominal and plastically deformed test assemblies. Plastically deformed assemblies will resemble assemblies with residual deformation due to bowing and dilation.

6.3.1.2.2 SCRAM Time/Impact Tests

This series of tests will examine control rod insertability and SCRAM time for the control rod system. Frictional behavior would be developed using component-level testing in air to build the model. This would be combined in the future with friction testing in sodium to predict performance in sodium. Withdrawal/insertion loads of the control rod bundle, frictional loads, and SCRAM time of both deformed and undeformed geometries in air will be measured for model benchmarking.

6.3.1.2.3 Control Assembly Static Load Deflection Tests

The purpose of this series of tests is to investigate and characterize the mechanical behavior of a control assembly. Tests will focus on mechanical behavior (bending stiffness, range of motion, etc.) of single assemblies in the elastic range as well as thermal effects.

6.3.1.2.4 Control Assembly Withdrawal & Insertion Tests

The purpose of this series of tests is to investigate and characterize the mechanical interactions of a control assembly with respect to core assembly handling loads. This includes removing both deformed and undeformed assemblies from variously configured clusters of neighboring assemblies within a test apparatus.

6.3.1.2.5 Control Assembly Seismic SCRAM Tests

This series of tests will examine control rod insertability and SCRAM time for the control rod system under a seismic event. Withdrawal/insertion loads of the control rod bundle, frictional loads, and SCRAM time of both deformed and undeformed geometries will be measured under various excitations for model benchmarking.

6.3.1.2.6 Single Assembly Free and Forced Vibration Tests

The purpose of this series of tests is to examine fundamental dynamic characteristics of Natrium core assemblies. This includes parameters such as natural frequency and structural damping.

6.3.1.2.7 Single Assembly Pluck Impact Test

The purpose of this test is to characterize dynamic impact behavior between assemblies at the top and above core load pads to calibrate contact behavior in seismic models.

The purpose of this test is to obtain the dynamic crush strength of the duct and load pads for use as a limit in seismic analyses. This would prevent damage to core assembly pin bundles.

6.3.1.3 Multiple Assembly Tests

6.3.1.3.1 Multiple Assembly Load Deflection Tests

The purpose of this series of tests is to investigate and characterize the mechanical behavior of SFR core assemblies with a focus on phenomena relevant to core restraint system design. Tests will focus on mechanical behavior (bending stiffness, range of motion, contact, etc.) of assemblies in the elastic range as well as thermal effects. Testing will include single assemblies as well as arrays of multiple assemblies.

6.3.1.3.2 Multiple Assembly Row and [[ ]]$^{(a)(4)}$ Cluster Seismic Tests

This series of tests will involve multiple assemblies in row or clustered configurations which will be subjected to excitation. The highly nonlinear response of the system due to inter-assembly gaps will be characterized using assembly displacement histories and contact/reaction loads and used for model benchmarking.

6.3.1.3.3 Multi-Assembly Withdrawal/Insertion Tests

The purpose of this series of tests is to investigate and characterize the mechanical interactions of SFR core assemblies with respect to core assembly handling loads. This includes removing both deformed and undeformed assemblies from variously configured clusters of neighboring assemblies within a test apparatus. The testing report will include data such as withdrawal, insertion, contact, and reaction loads.

6.3.1.4 Design Proof of Concept Tests

6.3.1.4.1 CRD/CRA Flow Induced Vibration

The purpose of this test is to evaluate the flow-induced vibration characteristics of the CRD and CRA under various flow conditions, and to evaluate fretting wear characteristics.

6.3.1.4.2 Control Rod Assembly Design Proof Testing

The purpose of this test is to evaluate the function of the control assembly/driveline coupling when subjected to misalignment and demonstrate the design intent.

6.3.1.4.3 Core Inlet Design Proof Test

The purpose of this test is to examine nozzle and receptacle fit at min/max material conditions, and to quantify wear on each component. Furthermore, mechanical hold-down force and drainage will be tested as well.

6.3.1.4.4 Lead Driver Assembly Design Proof and Duct Joint Load Testing

The purpose of this test is to test the connection of the lead driver assembly pins for removal, duct disassembly and reconstruction, duct joint design verification, and duct joint load limit identification.

6.3.1.4.5 Pin Wire Wrap Mechanical Test

The purpose of this test is to determine the structural integrity of the wire wrap weld and bend regions under static and dynamic loading conditions.

6.3.1.4.6 Orifice Plate Structural Integrity Test

The purpose of this test is to determine the structural integrity of the orifice plate under static and cyclic (fatigue) loading conditions.

6.3.1.4.7 Dashram/Dashpot Deceleration Test

The purpose of this test is to determine the spring/damper dynamic performance of the control assembly dashpot.

6.3.1.4.8 Vertical Drop to Base Test

The purpose of this test is to develop a data set that can be used to validate FEA models of a core assembly drop to the receptacle during refueling operations, including dynamic impact behavior.

6.3.1.4.9 Vertical Drop to Grapple

The purpose of this test is to develop a data set that can be used to validate FEA models of an impact between the core assembly and the IVTM grapple fingers, including impact force and position time histories.

6.3.1.5 Major Effects on Single and Multiple Fuel Assembly Test

The major effects identified for fuel and core assembly behavior are thermal gradient effects (TE), irradiation effects (IE), and fixity effects (FE), as summarized in Table 6-18. These effects must be accounted for when assessing assembly behavior. A notional test matrix is provided in Table 6-19 summarizing all of the envisioned tests for each of the major test categories, including the applicable major effects. Pending additional design and analysis effort the test matrix will be updated to ensure all high-importance phenomena with unknown and partial knowledge levels identified in the PIRTs [42] are adequately addressed.

Table 6-18. 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

Table 6-19. Example Fuel & Control Assembly Mechanical Test Matrix

(a)(4)

Type		Test	BOL	TE	IE	FE	Critical Characteristics
SA							
MA							
SC							
CA							

SA: Single Assembly MA: Multiple Assembly SR: Single Row
TE: Thermal Gradient Effects SC: Sub-Component
FE: Fixity Effects (Normal, Loose, Tight) CA: Control Assembly BOL: Beginning-Of-Life
IE: Irradiation effects accounted for by using pre-deformed ducts to simulate dilation and/or bowing induced by irradiation creep and growth. A hydraulic forming technique and device have been developed for duct deformation. Note that irradiation effects should be treated conservatively and will be by validated by surveillance programs.

6.3.2 Materials Property Data and Testing

The scope of qualification of materials property data includes the materials selected for each of the fuel components, as well as hard-face coatings and weldments applied to them.

Several handbooks compiling materials property data of HT9 for design input have been produced by US national laboratories, including the Nuclear Systems Materials Handbook (NSMH) [118], the Fuel Cycle Research and Development (FCRD) Materials Handbook under the Advanced Fuel Cycle Initiative (AFCI) [18], and the Generation IV Materials Handbook [119]. Similar compilations have been performed for metallic fuels, including the Metallic Fuels

Handbook [120], Thermophysical Properties of Matter Data Series [121], Metallic Fuels Handbook, Part 1 [122], and the Current revision of the Metallic Fuels Handbook summarizing properties of fresh fuels [123]. These handbooks contain a significant volume of information needed for design and analysis.

Additionally, TerraPower is reviewing literature references reporting on testing and characterization of the selected materials, as appropriate. These references are being catalogued in databases to determine their utility to contribute information for determination of physical/thermophysical, mechanical, and corrosion properties, as well as microstructure.

The qualification of legacy data will be performed following established procedures involving the four methods previously outlined in Section 6.2.1. Qualified data and applicable correlations will be formally documented in the Natrium Materials Handbook, an update to the current TerraPower Materials Handbook. Metallic fuel material (i.e., U-10Zr) properties data and applicable correlations will be documented in the Fuel Material Properties Handbook.

The method of qualification of materials data will depend on the availability of an existing database, availability of additional testing capabilities, and the safety function of the component for which the material data is a critical characteristic. Design inputs that take values directly from NRC or NNSA accepted standards such as ASME Boiler Pressure Vessel Code are considered pre-qualified by quality assurance program equivalency. Additional details are provided in the following paragraphs.

Qualification of material properties data by corroboration will rely on experimental data from multiple sources to demonstrate sufficient justification in the best estimate and uncertainty material property values that are used as design input for the fuel design. The data used must be properly referenced and verified by an independent reviewer before being accepted for qualification. The data for qualification by corroboration shall be taken from at least two or more references at minimum to meet the criteria of sufficient quantity.

Qualification of materials properties data by confirmatory testing will rely on well controlled and well documented testing performed in compliance with the TerraPower QA program. The testing may be subcontracted to another vendor with a 10 CFR 50 Appendix B (Appendix B) or equivalent quality assurance program or be commercially dedicated by TerraPower as an approved vendor for that specific testing service. Vendors who have Appendix B equivalent quality assurance programs must undergo TerraPower audits. Vendors without Appendix B equivalent programs may still provide the specific confirmatory testing service after a commercial grade dedication by TerraPower. The data obtained by confirmatory tests must be reviewed and accepted by TerraPower.

Qualification of materials properties data by peer review will rely on the expertise and professional judgement of a review team. The review team will be made up of a minimum two peer review members with relevant technical experience or expertise relating to the test method, history, standard, and/or analysis relating to the data. A peer review plan will be drafted in addition to the qualification plan by the responsible engineer to define the scope of the data review and provide all relevant supporting information to the review team prior to official review activities.

Qualification of materials properties data may be achieved by evaluating the quality program under which it was acquired and making positive correlations between that quality program and Appendix B. An example would be qualification of reactor materials data acquired under DOE programs.

Once the data has been qualified, the data will be labelled as “controlled” such that the information can be accessed as qualified data and accepted for use as design inputs.

6.3.2.1 HT9 Data for Qualification

Specifically, for HT9, this section identifies the required materials properties data that will impact the RAC under current TerraPower scope, and the intended methods for qualification for each dataset. The qualification of existing data procedure [76] will be used to qualify existing materials data that act as design input for the Natrium Reactor. A total of 21 qualification of existing data documents are identified for HT9 to support RAC associated with the reactor core. For organization and prioritization, the 21 qualification of existing data documents are binned into four property types: thermal properties, unirradiated mechanical properties, irradiated mechanical properties, and chemical interactions.

Table 6-20. Summary of HT9 Data Qualification

Property Type	Data Type	Associated RAC	Qualification Method
Thermal	Thermal expansion coefficient	4.2-1.1, 4.3-7	[[ ]](a)(4)
	Heat capacity	4.2-2.1, 4.2-3.2	[[ ]](a)(4)
	Thermal conductivity	4.2-2.1, 4.2-3.2	[[ ]](a)(4)
	Melting point	4.2-2.1, 4.2-2.3, 4.2-2.5, 4.2-3.2	[[ ]](a)(4)
Unirradiated mechanical	Young’s Modulus (unirradiated)	4.2-1.1, 4.2-3.1, 4.2-3.3, 4.2-3.5, 4.2-4.1	[[ ]](a)(4)
	Yield strength (unirradiated)	4.2-1.1, 4.2-1.8, 4.2-2.3, 4.2-2.4, 4.2-3.5	[[ ]](a)(4)
	Fracture toughness (unirradiated)	4.2-1.1, 4.2-1.8	[[ ]](a)(4)
	Fatigue (unirradiated)	4.2-1.2	[[ ]](a)(4)
	Creep (unirradiated)	4.2-1.1, 4.2-1.2, 4.2-1.6, 4.2-1.7, 4.2-2.1, 4.2-2.3, 4.2-3.1, 4.2-3.3, 4.2-3.5, 4.2-4.1	[[ ]](a)(4)
	Thermal aging	4.2-1.1, 4.2-1.8, 4.2-2.3, 4.2-3.5, 4.2-4.1	[[ ]](a)(4)

Property Type	Data Type	Associated RAC	Qualification Method
	Friction and wear	4.2-1.3, 4.2-2.5	[[ ]](a)(4)
Irradiated mechanical	Irradiated yield strength	4.2-1.1, 4.2-1.8, 4.2-2.3, 4.2-2.4, 4.2-3.5	[[ ]](a)(4)
	Irradiated fracture toughness	4.2-1.1 4.2-1.8	[[ ]](a)(4)
	Irradiation fatigue	4.2-1.2	[[ ]](a)(4)
	Irradiation creep	4.2-1.1, 4.2-1.2, 4.2-1.6, 4.2-1.7, 4.2-2.3, 4.2-3.1, 4.2-3.3,4.2-3.5, 4.2-4.1	[[ ]](a)(4)
	Irradiation swelling	4.2-1.1,4.2-1.6, 4.2-1.7, 4.2-3.1, 4.2-3.3, 4.2-3.5, 4.2-4.1	[[ ]](a)(4)
	Stress enhanced irradiation swelling	4.2-1.1, 4.2-3.1, 4.2-3.3, 4.2-3.5, 4.2-4.1	[[ ]](a)(4)
Chemical interactions	Cladding coolant compatibility (Sodium corrosion and erosion)	4.2-1.4, 4.2-2.5	[[ ]](a)(4)
	Cladding fuel compatibility	4.2-1.5, 4.2-2.3, 4.2-2.5	[[ ]](a)(4)
	Cladding absorber compatibility	4.2-1.7, 4.2-1.8	[[ ]](a)(4)
	Cladding reflector compatibility	4.2-1.6	[[ ]](a)(4)

6.3.2.1.1 Thermal Properties

Four HT9 thermal properties have been identified as design inputs for demonstrating the safety function of safety-significant components in the Natrium design. They must be qualified to demonstrate regulatory compliance.

The design inputs are melting point, coefficient of thermal expansion, heat capacity and thermal conductivity. All the properties have been measured historically.

Thermal properties for steels are well understood and mainly depend on the atomic bonding strength of the crystal lattice of the bulk material. Therefore, the thermal properties should not exhibit significant heat to heat variation. This widens the existing thermal properties database to allow inclusion of any body-center cubic (BCC) FM steel with nominal composition of 12Cr-1MoVW to be used as corroborative data for TerraPower HT9.

6.3.2.1.2 Unirradiated Mechanical Properties

Seven HT9 unirradiated mechanical properties have been identified as design inputs for demonstrating the safety function of safety-significant components in the Natrium Reactor design: Young's modulus, yield strength, fracture toughness, thermal creep, fatigue, thermal aging, friction and wear.

All these properties will have existing data that serve as design input that require qualification and can be confirmed by independent testing on TerraPower HT9.

Mechanical properties for steels are well understood and they can be highly variable depending on the final microstructure of the commercial product. Unlike thermal properties, it is generally not appropriate to directly use mechanical properties of steels that are not within the known HT9 specification to act as corroborative data for TerraPower HT9. In special cases where there is significant lack of existing data, technical justification must be made for the inclusion of data from materials outside of HT9's specification as part of the qualification of existing data process [124].

If needed, confirmatory tests could be conducted on TerraPower HT9 in addition to existing data. Any confirmatory tests conducted on TerraPower HT9 shall follow 10 CFR 50 Appendix B quality assurance requirements, and the test methods employed will be conducted in accordance with existing standards where appropriate.

6.3.2.1.3 Irradiated Mechanical Properties

Six HT9 irradiated mechanical properties have been identified as design inputs for demonstrating the safety function of safety-significant components in the Natrium Reactor: irradiated yield strength, irradiated fracture toughness, irradiation creep, irradiation swelling, stress enhanced swelling, and irradiation fatigue. Some of those properties have been measured historically and others have not.

TerraPower is conducting mechanical testing of HT9 materials irradiated in sodium-cooled fast neutron test reactors to relevant exposures. However, due to the limitation of existing prototypical data and difficulty in conducting mechanical tests on irradiated samples, advanced testing methods and analysis techniques may be included as corroborative data to support assumptions and confirm expected materials behavior.

It is recognized that almost no confirmatory testing on irradiated material will be fully prototypical. Therefore, the purpose of qualification of existing data and the performance of relevant confirmatory testing is to establish reasonable assurance that HT9 performance under irradiated conditions is within the design margins of the Natrium Reactor. Any confirmatory tests conducted on TerraPower HT9 shall follow 10 CFR 50 Appendix B quality assurance requirements, and the test methods employed will be conducted in accordance with existing standards where appropriate.

6.3.2.1.4 Chemical Interactions

Three HT9 chemical interactions have been identified as design inputs for demonstrating the safety function of safety-significant components in the Natrium Reactor design: cladding coolant compatibility, cladding-fuel compatibility, cladding-absorber compatibility. Some of those properties have been measured historically and others have not.

TerraPower is conducting testing of prototypical fuel pins that have been irradiated in test reactors, which provides results relevant to chemical interactions. However, due to the limitation of existing prototypical data and difficulty in conducting chemical interaction tests on irradiated samples, advanced testing methods and analysis techniques may be included as corroborative data to support assumptions and confirm expected materials behavior.

It is recognized that almost no confirmatory testing on irradiated material will be fully prototypical and the purpose of qualification of existing data is to establish reasonable assurance that HT9 performance under irradiated conditions are within the design margins of the Natrium Reactor. Any confirmatory tests conducted on TerraPower HT9 will follow 10 CFR 50 Appendix B quality assurance requirements, and the test methods employed will be conducted in accordance with existing standards where appropriate.

6.3.2.1.5 Availability of HT9 Materials Property Data

With respect to the properties discussed above, the table below summarizes data availability from different sources, with a qualitative assessment of gaps.

Table 6-21. Availability of Data for HT9

Property	Data Source	Availability
Fundamental Physical and Thermophysical Properties		
Coefficient of Thermal Expansion (CTE)	TerraPower Materials Handbook, Open Literature	[[ ]](a)(4)
Density	TerraPower Materials Handbook, Open Literature	[[ ]](a)(4)
Specific Heat Capacity	TerraPower Materials Handbook, Open Literature	[[ ]](a)(4)
Thermal Conductivity	TerraPower Materials Handbook, Open Literature	[[ ]](a)(4)
Time-Temperature Dependence of Structure and Phases	FCRD Handbook, Open Literature	[[ ]](a)(4)
Melting Point	FCRD Handbook, Open Literature	[[ ]](a)(4)
Emissivity	FCRD and NSMH Handbooks, Open Literature	[[ ]](a)(4)

Property	Data Source	Availability
Fundamental Physical and Thermophysical Properties		
Electrical Resistivity	FCRD Handbook	[[ ]](a)(4)
Mechanical Properties		
Young's Modulus	TerraPower and FCRD Handbooks, Open Literature	[[ ]](a)(4)
Shear Modulus	FCRD Handbook, Open Literature	[[ ]](a)(4)
Poisson's Ratio	TerraPower and FCRD Handbooks, Open Literature	[[ ]](a)(4)
Yield Strength (YS)	TerraPower, FCRD, and NSMH Handbooks, Open Literature	[[ ]](a)(4)
Ultimate Tensile Strength (UTS)	TerraPower, FCRD, and NSMH Handbooks, Open Literature	[[ ]](a)(4)
Tensile Stress-Strain Curves	ACFI Handbook, Open Literature	[[ ]](a)(4)
Uniform Elongation	FCRD and NSMH Handbooks, Open Literature	[[ ]](a)(4)
Total Elongation	FCRD and NSMH Handbooks, Open Literature	[[ ]](a)(4)
Fracture Toughness	FCRD and NSMH Handbooks, Open Literature	[[ ]](a)(4)
Ductile-to-Brittle Transition Temperature	FCRD and NSMH Handbooks, Open Literature	[[ ]](a)(4)
Creep	FCRD and NSMH Handbooks, TerraPower Constitutive Model Report, Open Literature	[[ ]](a)(4)
Stress-Rupture of Pressurized Tubes	FCRD and NSMH Handbooks, Open Literature	[[ ]](a)(4)
Fatigue	FCRD and NSMH Handbooks, Open Literature	[[ ]](a)(4)

Property	Data Source	Availability
Fundamental Physical and Thermophysical Properties		
Creep Fatigue	FCRD and NSMH Handbooks, Open Literature	[[ ]]$^{(a)(4)}$
Wear Rate	N/A	[[ ]]$^{(a)(4)}$
Coefficient of Friction	N/A	[[ ]]$^{(a)(4)}$
Irradiation Effects		
On all other properties (especially hardening, such as yield strength, fracture toughness, DBTT, irradiation creep, etc.)	FCRD handbook, and open literature	[[ ]]$^{(a)(4)}$
On Structure and Phases	FCRD handbook, and open literature	[[ ]]$^{(a)(4)}$
Void Swelling	FCRD handbook, and open literature	[[ ]]$^{(a)(4)}$
Environmental Compatibility		
Fuel-Cladding Chemical Interaction (FCCI)	TerraPower planned testing	[[ ]]$^{(a)(4)}$
Absorber-Cladding Chemical Interaction (ACCI)	TerraPower planned testing	[[ ]]$^{(a)(4)}$
Compatibility with Sodium (Corrosion, Erosion)	TerraPower planned testing	[[ ]]$^{(a)(4)}$

6.3.2.2 Coatings and Weldments

Once selection of hard-face coatings and development of welding processes is completed, all the relevant properties for those features will be qualified using a similar methodology to that presented above for HT9 steel components.

6.3.2.3 Metallic Fuel Properties Data for Qualification

This section identifies the required material properties data that are needed to support design and analysis activities. Except for fuel thermal conductivity, none of the fuel material properties were identified as high-importance phenomena in the PIRT assessments relative to assessing fuel pin design criteria. In spite of the other fuel properties being of less direct importance for evaluating fuel pin design criteria, other analyses are heavily dependent on these properties (i.e., neutronics assessments) and having reliable materials property data is essential to characterizing the beginning of life conditions of the fuel system. The applicable fresh fuel properties of interest and the planned qualification method are summarized in Table 6-22. Irradiated fuel material properties are assessed by the fuel performance modeling tools, with detail provided in Section 6.4.1. [76]

Table 6-22. Summary of U-10Zr Material Properties Data to be Qualified

Property Type	Data Type	Dependencies/Details	Qualification Method
Thermal	Density	• Zirconium concentration • Temperature	[[ ]](a)(4)
	Thermal expansion	• Linear thermal expansion • Mean coefficient of thermal expansion	[[ ]](a)(4)
	Heat capacity	• Temperature • Zirconium concentration • Includes enthalpies of phase transformation including enthalpy of fusion	[[ ]](a)(4)
	Thermal conductivity	• Zirconium concentration • Temperature • Porosity • Sodium infiltration • Fission products	[[ ]](a)(4)
	Melting point	• Solidus temperature • Liquidus temperature • Zirconium concentration • Plutonium concentration	[[ ]](a)(4)
Unirradiated mechanical	Young's Modulus (unirradiated)	• Zirconium concentration • Temperature	[[ ]](a)(4)
	Thermal Creep (unirradiated)	• Zirconium concentration • Temperature	[[ ]](a)(4)

6.4 Analytical Predictions

Analytical predictions are a component of the overall safety assessment of the reactor. Fuel performance models are used to evaluate fuel design criteria under normal operation and accident scenarios. This evaluation may be used, for example, to determine whether the fuel pin is damaged, to quantify the number of fuel pin failures, or to determine if fuel melting has occurred. Fuel performance models rely on inputs from other tools and methodologies to set boundary conditions and field variables such as cladding surface temperature, power, burnup, DPA or fluence, and the spatial and temporal variations of these parameters. Similarly, performance models of control, shield, and reflector pins are used to evaluate design criteria for those components. While discussion of analysis methodologies is outside the scope of this document, information is provided here regarding fuel performance models to inform RAC related to design evaluations of fuel pins. Section 6.4.1 provides a high-level overview of the performance models used to assess pin design criteria and the associated tools or methodologies used for key inputs. To illustrate the planned approach to address AQAF Appendix A Evaluation Model (EM) Goals, Table 6-28 maps AQAF EM goals and key phenomena to tools and methodologies.

Analytical predictions of core assembly mechanical behavior are important to ensuring compliance with safety requirements and specific design criteria. Finite element models are generated using TerraPower developed pre-processing software (OXBOW) to automate mesh generation and application of loads and boundary conditions in accordance with established methods. Section 6.4.2 summarizes these methods and introduces software tools used for conducting analysis.

The TerraPower Engineering Design and Analysis Software Management Procedure [125] provides the software quality assurance (SQA) work activities required for the planning, acquisition, development, operation, maintenance, and retirement of evaluation models. This procedure is being followed for the development and use of the modeling tools described in the following sub-sections helping ensure the quality of the final evaluation models. Three different types of documents are created as part of this procedure (for internally developed software) to help ensure "The evaluation model contains the appropriate modelling capabilities [EM G1]": 1) Software Requirements Specification Documents are prepared in advance of software development to ensure all of the software requirements (including appropriate geometries, materials, and physics) are identified and independently reviewed, 2) Software Design and Implementation Documents are prepared to document how these requirements are met and implemented in the code, and 3) Software Test Reports are issued to demonstrate that the required physics and requirements are properly implemented. Benchmark Comparison Reports and Test Reports are the main documents generated by this procedure to help ensure "The evaluation model has been adequately assessed against experimental data [EM G2]." Additional guidance has been developed for the Natrium project for performing software verification and validation [126], as well for methodology development and assessment [127].

6.4.1 Pin Performance Models

Two independent fuel performance models (ALCHEMY and [[ ]]^(a)(4)) have been matured for the purpose of predicting fuel pin damage and failure for use in the core design and safety analysis methodologies. These two models were developed independently, utilize different numerical methods, and differ in the approaches to modeling certain phenomena. This allows for independent verification of analysis predictions and diversity in evaluation approaches. In addition, a third model (CRUCIBLE) is used for capturing changes to fuel pins during normal operation, which impact neutronic and thermal hydraulic calculations. The ALCHEMY model is also capable of modeling control, shield, reflector, and Type 1B pin performance. The following sections describe these models and their capabilities. The use cases for each model are summarized in Table 6-23. A mapping of applicable fuel performance models or methodologies to high-importance phenomena used to assess damage, failure, coolability, and fuel melting is provided in Table 6-24, Table 6-25,Table 6-26, and Table 6-27, respectively. FQAF goals related to fuel performance model are addressed at a high-level in Table 6-28.

Table 6-23. Applicable Models and Codes for Fuel Pin Phenomena

Use Case	Applicable Tool(s)
Ability to meet fuel pin design limits during normal operation to preclude fuel pin damage. Pre-transient characterization of fuel condition.	[[ ]]^(a)(4)
Prediction of fuel pin damage and failure during accidents.	[[ ]]^(a)(4)
Impact of fuel performance phenomena on normal operation neutronic and thermal hydraulic behaviors (e.g., fuel axial growth impact on core fuel density distribution).	[[ ]]^(a)(4)
Impact of fuel performance phenomena on transient conditions during accident (e.g., fuel axial growth impact on core fuel density distribution).	[[ ]]^(a)(4)

Use Case	Applicable Tool(s)
Ability to meet control, shield, and reflector pin performance limits during normal operation and postulated accidents.	[[ ]](a)(4)

Table 6-24. Fuel Performance Prediction Capabilities to Assess Fuel Damage

Applicable Design Limit	Applicable RAC	High-Importance Phenomena/ Parameters	Applicable Tool/Methodology
Total Peak Cladding Strain	4.2-1.1, 4.2-1.6, 4.2-1.8	Fission gas release	[[ ]](a)(4)
		HT9 mechanical response as a function of temperature, stress, irradiation, and time	[[ ]](a)(4)
		FCCI	[[ ]](a)(4)
		Fuel burnup	[[ ]](a)(4)
		DPA on cladding	[[ ]](a)(4)
		Cladding temperatures	[[ ]](a)(4)
Cladding Fatigue Lifetime	4.2-1.2	HT9 mechanical response as a function of temperature, stress, irradiation, and time	[[ ]](a)(4)
		Fuel burnup	[[ ]](a)(4)
		DPA on cladding	[[ ]](a)(4)
		Cladding temperatures	[[ ]](a)(4)
		Number of strain cycles on cladding	[[ ]](a)(4)
		Magnitude of strain cycles	[[ ]](a)(4)
Cladding Wastage	4.2-1.3, 4.2-1.4, 4.2-1.5	FCCI	[[ ]](a)(4)
		Cladding temperatures	[[ ]](a)(4)
		Residence time	[[ ]](a)(4)

Table 6-25. Fuel Performance Prediction Capabilities to Assess Fuel Failure

Applicable Design Limit	Applicable RAC	High-Importance Phenomena/ Parameters	Applicable Tool/Methodology
Peak Cladding Temperature	4.2-2.1	Detailed pin level irradiation histories including power and cladding temperature	[[ ]](a)(4)
		Detailed coolant transient temperature and pin power histories	[[ ]](a)(4)
	4.2-2.2	Fuel thermal conductivity	[[ ]](a)(4)

Applicable Design Limit	Applicable RAC	High-Importance Phenomena/ Parameters	Applicable Tool/Methodology
Peak Fuel Temperature		Detailed pin level irradiation histories including power and cladding temperature	[[ ]](a)(4)
		Detailed coolant transient temperature and pin power histories	[[ ]](a)(4)
Peak Cladding Thermal Creep Strain	4.2-2.3	Fission gas release	[[ ]](a)(4)
		HT9 mechanical response as a function of temperature, stress, irradiation, and time	[[ ]](a)(4)
		FCCI	[[ ]](a)(4)
		Fuel burnup	[[ ]](a)(4)
		Cladding temperatures	[[ ]](a)(4)
Cladding Wastage	4.2-2.5	FCCI	[[ ]](a)(4)
		Cladding temperatures	[[ ]](a)(4)
		Residence time	[[ ]](a)(4)

Table 6-26. Fuel Performance Prediction Capabilities to Assess Fuel Coolability

Applicable Design Limit	Applicable RAC	High-Importance Phenomena/ Parameters	Applicable Tool/Methodology
Total Peak Cladding Strain	4.2-3.1, 4.2-3.3, 4.2-3.5	Fission gas release	[[ ]](a)(4)
		HT9 mechanical response as a function of temperature, stress, irradiation, and time	[[ ]](a)(4)
		Fuel burnup	[[ ]](a)(4)
		Cladding temperatures	[[ ]](a)(4)
Peak Cladding Temperature	4.2-3.2	Detailed pin level irradiation histories including power and cladding temperature	[[ ]](a)(4)
		Detailed coolant transient temperature and pin power histories	[[ ]](a)(4)
Peak Fuel Temperature	4.2-3.4	Fuel thermal conductivity	[[ ]](a)(4)
		Detailed pin level irradiation histories including power and cladding temperature	[[ ]](a)(4)

Applicable Design Limit	Applicable RAC	High-Importance Phenomena/ Parameters	Applicable Tool/Methodology
		Detailed coolant transient temperature and pin power histories	[[ ]](a)(4)

Table 6-27. Fuel Performance Prediction Capabilities to Assess Phenomena Related to Fuel Temperatures

Phenomena/Parameters	Applicable Tool/Methodology
Radial power distribution due to constituent redistribution	[[ ]](a)(4)
Fuel temperature distribution	[[ ]](a)(4)
Cladding temperature distribution	[[ ]](a)(4)
Burnup distribution in the fuel	[[ ]](a)(4)
Thermal conductivity of the fuel and cladding	[[ ]](a)(4)
Thermal expansion of the fuel and cladding	[[ ]](a)(4)
Fission gas production and release	[[ ]](a)(4)
Solid and gaseous fission product swelling	[[ ]](a)(4)
Fuel deformation	[[ ]](a)(4)
Diffusion of fuel constituents	[[ ]](a)(4)
Fuel and cladding dimensional changes	[[ ]](a)(4)
Fuel-to-cladding heat transfer	[[ ]](a)(4)
Fuel-to-cladding contact pressure	[[ ]](a)(4)
Heat capacity of the fuel and cladding	[[ ]](a)(4)
Swelling and creep of the cladding	[[ ]](a)(4)
Rod internal gas pressure	[[ ]](a)(4)
Rod internal gas composition	[[ ]](a)(4)
Cladding-to-coolant heat transfer coefficient	[[ ]](a)(4)
Cladding wastage (erosion, corrosion)	[[ ]](a)(4)
FCCI	[[ ]](a)(4)

Table 6-28. Fuel Performance Models for FQAF Goals

FQAF Goal ID	Evaluation Model (EM) Assessment Framework Goal Description	Natrium Fuel Performance Modeling Approach
EM G1	Evaluation model contains the appropriate modeling capabilities	Model requirements have been informed by design criteria and PIRT analysis of important phenomena necessary for the prediction of design criteria. Evaluation models will demonstrate appropriate capabilities have been implemented through software testing.
EM G1.1	Evaluation model is capable of modeling the geometry of the fuel system	Pin performance models typically utilize an axisymmetric (RZ) dimensionality that

FQAF Goal ID	Evaluation Model (EM) Assessment Framework Goal Description	Natrium Fuel Performance Modeling Approach
		includes the fuel (or B_4C), cladding, and plenum regions. The dependence of boundary conditions (e.g., cladding temperature) that vary along the circumference of the fuel pin are typically evaluated through sensitivity analysis. Specialized models of the fuel cross-section or other dimensionalities are occasionally used for unique assessment on a case-by-case basis.
EM G1.2	Evaluation model is capable of modeling the material properties of the fuel system	Fuel performance models include material models for U-10Zr and HT9 cladding. Additional material models are used for benchmarking to historical experiments that may include U-Pu-Zr fuels and other cladding materials. Pin performance models for control, shield, reflector, and Type 1B pins employ material models applicable to the Natrium design.
EM G1.3	Evaluation model is capable of modeling the physics relevant to fuel performance	PIRT assessments have been used to identify important phenomena necessary for fuel performance predictions. Code requirements specify the necessary physics for each model.
EM G2	Evaluation model has been adequately assessed against experimental data	Initial validation assessments have demonstrated the ability to predict experiment data related to key design criteria such as cladding strain and pin failure. Detailed validation plans and assessments are under development to demonstrate that validation assessment criteria have been met.
EM G2.1	Data used for assessment are appropriate (see ED Assessment Framework)	Qualification of fuel performance data is being performed to qualify existing experiment data from EBR-II and FFTF. New experiments are being performed under an appropriate quality program.
EM G2.2	Evaluation model is demonstrably able to predict fuel failure and degradation mechanisms over the test envelope	Experiments that resulted in fuel failure are being used to validate fuel performance models. These include a small set of failures that occurred during normal operation, transient overpower experiments, furnace experiences, and a variety of materials tests and sub-system experiments. New experiments are being designed to fill gaps in the experimental databases and results will subsequently be used for model validation.

FQAF Goal ID	Evaluation Model (EM) Assessment Framework Goal Description	Natrium Fuel Performance Modeling Approach
EM G2.2.1	Evaluation model error is quantified through assessment against experimental data	Validation plans include the determination of model error via assessment to experiment data.
EM G2.2.2	Evaluation model error is determined throughout the fuel performance envelope	Validation plans include the determination of model error via assessment to experiment data.
EM G2.2.3	Sparse data regions are justified	Justification of regions with sparse data will be provided.
EM G2.2.4	Evaluation model is restricted to use within its test envelope	Coverage of validation assessments will be reported in software test reports.

6.4.1.1 ALCHEMY

ALCHEMY is a thermo-mechanical model based on the finite element method capable of simulating fuel, absorber, shield, and reflector pin behavior within a reactor environment. ALCHEMY generates the geometry and mesh for the problem, applies boundary conditions, models nuclear-specific material behavior and phenomena, and solves the coupled thermo-mechanical equations describing the physics being simulated. It has been developed to work in conjunction with the commercial-off-the-shelf finite element analysis software ABAQUS.

ALCHEMY includes a preprocessor, postprocessor, user subroutines that work in conjunction with ABAQUS, and several features to assist in analysis, verification, and validation. The preprocessor takes model input variables from a user-supplied input file and implements the ABAQUS Application Programming Interface (API) to automatically create the geometry, mesh, and boundary conditions of the model. Model solutions occur through ABAQUS, in conjunction with ALCHEMY-CORE, a set of user subroutines to extend the ABAQUS functionality. The post-processor is used to obtain simulation data from solution files. The pre- and post-processing is written in Python, whereas the user subroutines are written in FORTRAN.

The software allows the user to specify [[
]](a)(4). Fuel composition is generally a mixture of uranium, plutonium, and zirconium while the cladding materials are typically a steel alloy. This flexibility in choice of materials and geometry allows for the simulation of legacy fuel pin designs and experiments as well as current fuel pin designs. In addition, the user can specify a variety of operating conditions for normal operation, start-ups, shutdowns, and accident scenarios. ALCHEMY also provides capabilities to simulate separate effects experiments of metallic fuel or structural materials.

The software also has the capability to model boron carbide (B_4C) in place of fuel, which can be used to simulate the behavior of Natrium absorber pins or shield pins. The Natrium reflector design ([[ ]](a)(4)) can also be modeled. Encapsulated experiments, such as those used in the ATR can be modeled by specifying the capsule geometry and associated boundary conditions.

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6.4.1.3 CRUCIBLE

CRUCIBLE is the primary tool for modeling the axial growth of the fuel, changes to the sodium bond, and fission gas release during normal operation. These parameters influence the composition of the core and geometry of the fuel, and thus influence the reactor normal operation neutronic and thermal hydraulic behaviors. These fuel performance phenomena are generally modeled using empirical correlations derived from data measured from historical metal fuel tests. Due to the observed variability of these phenomena, CRUCIBLE can apply uncertainty factors that vary the empirical models within the observed variability. Additionally, CRUCIBLE can accept user-defined inputs for these phenomena, which allow for further sensitivity studies to assess the impact to normal operation conditions.

CRUCIBLE provides an interface for calculating fuel temperatures under simplifying assumptions compared to the more detailed ALCHEMY and [[ ]](a)(4) models. This temperature calculation is used to determine the thermal expansion axial growth of the fuel and for assessing peak fuel temperatures considering uncertainties during normal operation.

CRUCIBLE is not expected to play a role in the evaluation of fuel pin design criteria, but is included here for completeness, and to highlight how these fuel performance phenomena are accounted for to support other core design activities.

6.4.2 Core Assembly Mechanical Analysis

Predicting core assembly behavior (particularly for fuel assemblies) is an important aspect of sodium fast reactor (SFR) core mechanical design. One aspect of this is how displacements of fuel within the neutron flux gradients due to assembly deformations change the reactivity of the core. These reactivity feedbacks have effects on reactor safety and operation. An additional concern relates to the amount of deformation assemblies accumulate during their residence time in the core which affects handling operations and assembly useable lifetime. The types of deformations that pertain to an SFR core assembly are listed in subsequent sections. All core assemblies will be analyzed for these distortions, but generally fuel assemblies are the most limiting since they experience the highest neutron doses and temperature gradients in the core. The mechanical performance and integrity of limiting assemblies may dictate shuffling operations or core management. Sections 6.4.2.1 through 6.4.2.6 summarize the core assembly phenomena of concern and the associated analysis methods and software used for conducting the corresponding mechanical analyses.

6.4.2.1 Core Assembly Distortion [128, 129]

Predicting core assembly behavior (particularly for fuel assemblies) is an important aspect of sodium fast reactor (SFR) core mechanical design. One aspect of this is how displacements of fuel within the neutron flux gradients due to assembly deformations change the reactivity of the core. These reactivity feedbacks have effects on reactor safety and operation. An additional concern relates to the amount of deformation assemblies accumulate during their residence time in the core which affects handling operations and assembly useable lifetime. The types of deformations that pertain to an SFR core assembly are listed in subsequent sections. All core assemblies will be analyzed for these distortions, but generally fuel assemblies are the most limiting since they experience the highest neutron doses and temperature gradients in the core. The mechanical performance and integrity of limiting assemblies may dictate shuffling operations or core management.

6.4.2.1.1 Core Restraint System [128, 129]

Evaluation of core-wide assembly bowing and interaction forces is important to predicting the performance of the Natrium core over its lifetime. As part of the CRS, the evolution of core assembly bowing and accumulated bowing deformations are used to predict the radial expansion reactivity feedback mechanism for a fresh core or a configuration that has undergone multiple cycles of operation. Inter-assembly interaction forces, which result from assembly bowing and dilation, inform handling loads for the various core assemblies as well as the loads that are reacted by the core support structures. The core restraint system is described in detail in Section 5.4.

6.4.2.1.2 OXBOW.CRS

The TerraPower-developed code OXBOW.CRS will be used for analysis of core-wide assembly behavior and core restraint system analysis. This code utilizes assembly load history information in order to evaluate the evolution of core-wide assembly bowing and interaction forces. OXBOW.CRS makes use of the finite element method utilizing the commercially available solver ABAQUS. The finite element model (FEM) is created using geometry, boundary conditions, and loads provided by other physics codes and core component design.

A sub-module of OXBOW.CRS (OXBOW.SHUFFLE) is used to account for multi-cycle effects and refueling, and incorporates inelastic deformations predicted using OXBOW.CADA, described below.

6.4.2.1.3 OXBOW.CADA

The TerraPower-developed code OXBOW.CADA will be used for core assembly distortion analysis. This code calculates duct dilation, duct axial growth, and core assembly bowing for a given load history. OXBOW.CADA makes use of the finite element method utilizing the commercially available solver ABAQUS. Custom creep and swelling user materials from the fuel performance code ALCHEMY are used in these ABAQUS models. These material subroutines incorporate thermal creep, irradiation creep, and void swelling correlations for stainless steels used in core design. The FEM is created using geometry, boundary conditions, and loads provided by other physics codes.

6.4.2.2 Core Seismic

The seismic response of the reactor core system is important to the overall performance of the Natrium Reactor. It is important to be able to predict the reactivity response as well as the structural response of the core during a seismic event. The Natrium core will be analyzed for seismic licensing basis events (LBEs) of various severities.

6.4.2.2.1 OXBOW.SEISMIC

The TerraPower-developed code OXBOW.SEISMIC will be used for analysis of core-wide assembly behavior under seismic loads. This code utilizes lateral seismic excitations in order to evaluate the evolution of core-wide assembly bowing and impact forces. OXBOW.SEISMIC makes use of the finite element method utilizing the commercially available solver ABAQUS. The FEM is created using geometry, boundary conditions, and loads provided by the core design team.

6.4.2.3 Core Assembly Withdrawal/Insertion [128, 129]

It is important to be able to quantify the magnitude of the withdrawal and insertion loads that occur when performing refueling operations. An understanding of the magnitude of loads required to handle core assemblies is necessary for efficient core management. The event of a core assembly not being able to be removed or inserted should be avoided as it will result in the interruption of normal reactor operations and require a very costly intervention. Because economical operation of the Natrium Reactor relies on significant amount of fuel shuffling this analysis is important for core design.

6.4.2.3.1 OXBOW.WI

TerraPower-developed code OXBOW.WI will be used for core assembly withdrawal and insertion analysis. This code calculates the frictional interactions as well as the handling loads required to withdraw and insert assemblies of various deformations from an array of neighbors. OXBOW.WI makes use of the finite element method utilizing the commercially available solver ABAQUS. The FEM is created using geometry, boundary conditions, and loads provided by the core design team. Deformed assembly state information can be generated using OXBOW.CRS and OXBOW.CADA.

6.4.2.4 Control Assembly SCRAM [128, 129]

Predicting the mechanical behavior of the control rods is an important part of core design. The amount of deformation that a Natrium control rod and control assembly accumulates over its lifetime may limit functionality. Understanding how much of their functionality is limited is a key effort to ensuring safe and reliable reactor operation. The scope of this analysis methodology is to assess the withdrawal and insertion mechanical response of control rods.

6.4.2.4.1 OXBOW.CASS

The TerraPower-developed code OXBOW.CASS will be used for analysis of control assembly function under various operating conditions. This code utilizes deformed control assembly state information to evaluate the insertion and withdrawal capability of the control rod bundle. OXBOW.CASS makes use of the finite element method utilizing the commercially available solver ABAQUS. The FEM is created using geometry, boundary conditions, and loads provided by the core design team. Control bundle insertability and SCRAM time analyses are conducted using OXBOW.CASS.

6.4.2.5 Core Assembly Pin Bundle/Duct Interaction [128, 129]

Over the lifetime of a core assembly, the assembly duct and the internal pin bundle will deform. The duct will bow and dilate while the pin bundle will swell. As these components deform at different rates, clearance or interference may develop at their interfaces. Excessive clearance may result in flow-induced vibration, fretting, and excessive coolant bypass as flow area is increased. Conversely, significant interference may result in mechanical interaction between a pin bundle and duct or a decrease in the flow area which increases the assembly pressure drop. Additionally, coolant flow restriction could cause overheating of fuel pins. The clearance or interference change resulting from differential duct and pin bundle deformations will be analyzed.

A similar type of analysis is needed for control assemblies. Absorber pins within a control rod bundle swell at a different rate than the control rod duct. Additionally, the control rod duct may experience different deformation modes than the control assembly duct. The unique clearance or interference of control assembly components will be calculated for comparison against design limits.

6.4.2.5.1 OXBOW.BDI

The TerraPower-developed code OXBOW.BDI will be used for analysis of bundle-duct interaction. This code calculates duct dilation, duct axial growth, core assembly bowing, and pin bundle deformation for a given load history and tracks pin positions resulting from external loading conditions on the core assembly. OXBOW.BDI makes use of the finite element method utilizing the commercially available solver ABAQUS. Custom creep and swelling user materials from the fuel performance code ALCHEMY are used in these ABAQUS models. These material subroutines incorporate thermal creep, irradiation creep, and void swelling correlations for stainless steels used in core design. The FEM is created using geometry, boundary conditions, and loads provided by the core design team.

6.4.2.6 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. The core assemblies will have a structural evaluation plan and analysis methodologies, which will rely on finite element analyses. Effects of irradiation will be accounted for. The core support structure will need to be evaluated to ASME Section III, Division 5, which will require structural analysis.

6.4.2.6.1 Core Assembly Drop while attached to Handling Machine Grapple

This drop could occur if a partially inserted core assembly were to contact adjacent core assemblies with sufficient force such that a push from the handling machine is required to overcome friction and continue insertion. If the contact force from the adjacent core assemblies were suddenly released once the above core load pad of the inserted assembly passes the top load pads of the adjacent assemblies, the inserted assembly could drop onto the grapple interface.

6.4.2.6.2 Core Assembly Drop Following Release from the Handling Machine Grapple

For a drop of this scenario to occur, the core assembly would have to be stuck at some elevation above its seated position and held by adjacent assemblies while the grapple is released and withdrawn. Then it subsequently would be released to drop into its receptacle position in the core support structure.

6.4.2.6.3 Core Assembly Drop Following Lift-off due to Vertical Seismic Loads

During an earthquake, core assemblies could be lifted off from their receptacle positions due to a high vertical load that exceeds the hold-down margin of the core assemblies. Depending on the dynamic response of the core assemblies and core support structures, there could be an out-of-phase displacement that causes a high impact load. Note that this impact load could be more severe than the above scenarios since the impact velocity could be higher than that of a free drop in sodium or push by the grapple.

7. TESTING AND INSPECTION OF NEW FUEL

RAC 4.2-7 identifies the expectation that “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.” The bulk of the required activities will be specified in the fuel system product specifications, but to help ensure all testing and inspections are adequately captured, Table 7-1 summarizes the specific needs identified in RAC 4.2-7 along with the anticipated approach to address the requirements.

Table 7-1. Summary of New Fuel Testing and Inspection Needs and Planned Approach

Requirement	Planned Approach to Address
Cladding integrity	Certificates of conformance and supporting quality documentation demonstrating compliance with product specification requirements. Use of qualified manufacturing processes and suppliers.
Fuel system dimensions	
Fuel enrichment and chemical composition	
Absorber composition	
Onsite inspection of new fuel and control assemblies to ensure delivered quality	Program for receipt inspection and acceptance of new fuel assemblies after delivery to the plant

8. ONLINE FUEL SYSTEM MONITORING FOR FUEL PIN FAILURE

Design and development of the online fuel monitoring system is addressed as part of the overall plant design effort and will be covered in more detail in future submittals. Only a brief summary is provided here for clarity on key points of fuel failure detection and identification. The primary method of determining if a fuel pin breach has occurred will be accomplished by continuously monitoring the cover gas effluent for the presence of radioactive fission gases as proven in FFTF. Upon gas plenum breach the gas is immediately released with the rate being [[

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9. FUEL SURVEILLANCE

Even though there is high confidence in Type 1 fuel being able to readily achieve its full design lifetime a comprehensive fuel surveillance program is planned to closely monitor fuel performance. The plan is to start operation with [[ ]](a)(4) LDAs in the core, with a number of LDAs removed after each cycle of operation to perform post-irradiation exams to verify the fuel is performing consistent with expectations. See Table 9-1 for the notional planned irradiation of these LDAs during early cycles of operation to support fuel surveillance.

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This report contains an assessment of data and testing required to support fuel pin licensing with particular focus on experiments supporting the fuel design limits/criteria, as well as high-importance phenomena that influence the ability to reliably meet design criteria. These high-importance phenomena will be monitored as part of the fuel surveillance program to verify consistent performance. Specifically, 1) visual exams will be performed to identify any potential signs of wear or corrosion, 2) cladding and duct dimensions will be monitored to verify the integral response of the fuel pins/assemblies and determine overall cladding strain, 3) neutron radiography to verify the amount of fuel axial growth, 4) fission gas release measurements, and 5) FCCI measurements.

The use of LDAs in the Natrium surveillance program will provide early indications of any potential off-normal behavior and will supplement available in-reactor data to further reduce uncertainties in the fuel performance models. Specific fuel performance uncertainties to be addressed by the Fuel Surveillance Program are summarized in Table 9-2.

Table 9-1. Notional Fuel Surveillance Plan for Initial Cycles of Operation

[[ ]](a)(4)	**Cycles in the Reactor**	**Total Number of Cycles in the Core**	**Targeted Cladding Temperature**	**Objective**
[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	
[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	
[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	

[[ ]](a)(4)	Cycles in the Reactor	Total Number of Cycles in the Core	Targeted Cladding Temperature	Objective
[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)
]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	[[ ]](a)(4)	
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Table 9-2. Fuel Performance Uncertainties and Mitigation Steps

Fuel Performance Uncertainty	Mitigation Step
[[ ]](a)(4)	[[ ]](a)(4)
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The final surveillance program will be dependent on the success of the other items previously identified in this report. Both non-destructive exams (NDE) and destructive exams (DE) to measure specific phenomenon are identified above to provide continued assurance of consistent fuel behavior. The target will be to have these exams performed in parallel with subsequent reactor cycles to prevent disruption of operation. [[ ]](a)(4)

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10. CONCLUSIONS

A systematic assessment was performed to identify the activities required to support fuel system qualification. In general, the ongoing activities appear adequate to address most of the qualification needs. A few key exceptions are efforts to address fretting and fatigue behavior, as well as additional testing and analysis to address extreme transients, including coolability concerns. Initiation of many of these activities were delayed because they require prototypic fuel bundle geometries and anticipated operating and design basis event conditions. Now that conceptual design information is available, detailed test planning and supporting fabrication activities are underway. The current qualification plan relies heavily on historic operating experience/data from EBR-II and FFTF metallic fuel pins, with an ongoing effort to qualify the existing data to demonstrate its suitability [66]. The planned data qualification approach is consistent with the Quality Assurance Program Plan [77] submitted by Argonne National Laboratory for review and approved by the NRC [78]. The high-importance fuel phenomena identified for applicable fuel pin design limits include fission gas release, HT9 mechanical behavior as a function of environmental conditions, FCCI, and fuel thermal conductivity as a function of irradiation/porosity. A comprehensive set of test and analysis activities to address the limitations in these phenomena, and strengthen the basis of the associated design criteria, is summarized in Table 6-11 through Table 6-17. A series of mechanical tests have been identified for the fuel and control assemblies to address uncertainties in their response and to support model validation. These tests include component tests, single assembly tests, multiple assembly tests, and major effects tests. A summary of these tests, including associated effects, is provided in Table 6-19. Evaluation models for analytic predictions are available and capable of addressing most of the high-importance phenomena, with ongoing development to address existing gaps in time to support submission of the Final Safety Analysis Report, including verification and validation of the methods.

With no available fast-spectrum reactor to perform final tests using prototypic LTAs, a notional Surveillance Program is proposed to help monitor the irradiation performance of the fuel to help ensure consistent performance with historic operating experience and analytical predictions. The notional Surveillance Program will be revised to incorporate knowledge gained from additional analyses and testing data that becomes available.

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