ENCLOSURE 5

Core Design and Thermal Hydraulic Technical Report

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ACRONYMS

Acronym	Definition
ACLP	Above Core Load Pad
ANL	Argonne National Laboratory
BOEC	Beginning of Equilibrium Cycle
BOFC	Beginning of First Cycle
BOL	Beginning of Life
CFD	Computational Fluid Dynamics
[[	]](a)(4)
CRBRP	Clinch River Breeder Reactor Plant
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
DHGR	Decay Heat Generation Rate
DPA	Displacements per Atom
EBR-II	Experimental Breeder Reactor-II
EFPD	Effective Full Power Days
EOEC	End of Equilibrium Cycle
FEA	Finite Element Analysis
FFTF	Fast Flux Test Facility
FIMA	Fissions per Initial Metal Atom
FOPT	First Order Perturbation Theory
HCF	Hot Channel Factor
HFG	Hyperfine Group
HFP	Hot Full Power
HZP	Hot Zero Power
IHT	Intermediate Heat Transport System
IHX	Intermediate Heat Exchanger
IVS	In-Vessel Storage
LBE	Licensing Basis Event
LWR	Light Water Reactor
MIT	Massachusetts Institute of Technology
MOEC	Middle of Equilibrium Cycle
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
[[	]](a)(4)
[[	]](a)(4)
PDC	Principal Design Criteria
PHT	Primary Heat Transport System
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PSP	Primary Sodium Pump

[[	]](a)(4)
RCC	Reactor Core System
RES	Reactor Enclosure System
RIL	Rod Insertion Limits
RV	Reactor Vessel
RZMFLX	Regular Zone-averaged Moment Flux
SARRDL	Specified Acceptable System Radionuclide Release Design Limits
SAS	SAS4A/SASSYS-1
SFR	Sodium Fast Reactor
SSCs	Structures, Systems, and Components
TWR	Travelling Wave Reactor
UFG	Ultrafine Group
UIS	Upper Internal Structure
V&V	Verification and Validation

1 INTRODUCTION

This report is developed in support of a construction permit application being submitted in accordance with 10 CFR 50.34(a), “Preliminary Safety Analysis Report,” for Kemmerer Power Station Unit 1 [1]. This report describes the core design and analysis methodology for the reactor at the beginning of life and at equilibrium conditions. This methodology is used as an appropriate means to develop and analyze the core nuclear and thermal hydraulic design for normal operation and downstream use in nuclear safety analysis for the Natrium™ reactor[1].

1.1 Scope

The purpose of this report is to summarize the steady-state nuclear, thermal hydraulic, and core mechanical modeling of the reactor core and multi-physics simulations capturing key phenomena affecting the nuclear and thermal hydraulic of the system, such as fuel burn-up expansion, fission gas release, uncertainty treatment in component temperatures, and the influence of mechanical interactions of core assemblies for reactivity characterization. The scope of this technical report supports the preliminary design of the reactor core and sample evaluations are provided based on the current design of the Reactor Core System (RCC). A summary description of each section is provided in Table 1-1.

Table 1-1: Technical Report Overview

Section	Description
Section 2– Reactor Design	Reactor design description and modeling approach applied for integrated nuclear and thermal hydraulics evaluations.
Section 3 – Codes and Methods	Description of nuclear, thermal hydraulics, and mechanical codes and methods used for evaluating steady state design characteristics of the reactor core.
Section 4 – Steady State Core Design Methodology	Analysis methodologies for evaluating key nuclear and thermal hydraulics figures of merit with uncertainty treatment of peak inner cladding temperature with hot channel factors.
Section 5 – Methodology Validation	Methodology assessment plans for nuclear, thermal hydraulic, and core assembly bowing evaluations with benchmark data.
Section 6 – Summary and Conclusions	Technical report summary.
Section 8 – Appendix Example Steady State Core Design Results	Steady state core design evaluations with applied methodologies.

Features of the reactor design and validation of the nuclear, thermal hydraulics, and mechanical codes and methods are preliminary at this time and noted throughout this technical report.

[1] a TerraPower & GE-Hitachi Technology

1.2 Regulatory Requirements

Nuclear Regulatory Commission (NRC) regulations in 10 CFR 50.34(a)(4) requires an analysis and evaluation of the design and performance of Structures, Systems, and Components (SSCs) of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

The methodologies described in this report are used to analyze the core during normal operation, and to provide analytical outputs for transient and safety evaluations of the Natrium reactor.

1.2.1 Principal Design Criteria

The Kemmerer Unit 1 Preliminary Safety Analysis Report (PSAR) provides a set of principal design criteria (PDC) developed for and informed by design features specific to the Natrium reactor. The analytical models and example results presented in this report support PSAR discussions on implementation of the PDC listed below.

Specific to PDC 12, the method for evaluating power oscillations that do not result in conditions exceeding Specified Acceptable System Radionuclide Release Design Limits (SARRDLs) is described in TP-LIC-RPT-0006 “Stability Methodology Topical Report” [2], and is not included in this report. However, several key inputs required in applying that evaluation to the Natrium design are provided from the methods summarized in this report, such as neutron generation time and core-wide reactivity coefficients.

Additional detail on the design’s implementation of the PDC listed below is discussed in Section 5.3 of the Kemmerer Unit 1 PSAR [1].

PDC 10, Reactor Design

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

PDC 11, Reactor Inherent Protection

The reactor core and associated systems that contribute to reactivity feedback are designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

PDC 12, Suppression of Reactor Power Oscillations

The reactor core; associated structures; and associated coolant, control, and protection systems are designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.

PDC 26, Reactivity Control Systems

The nuclear design analysis is performed to confirm that minimum of two reactivity control systems or means provides:

1. *A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the specified acceptable system radionuclide release design limits are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences.*
2. *A means which is independent and diverse from the others, is capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the specified acceptable system radionuclide release design limits are not exceeded.*
3. *A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.*
4. *A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection, and repair shall be provided.*

PDC 28, Reactivity limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the safety-significant elements of the primary coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor vessel internals to impair significantly the capability to cool the core.

PDC 29 - Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

PDC 34, Residual heat removal

A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable system radionuclide release design limits and the design conditions of the primary coolant boundary are not exceeded.

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

PDC 80, Reactor Vessel and Reactor System Structural Design Basis

The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.

1.3 Design Features

The Natrium reactor is a pool-type, sodium-cooled, fast reactor which operates at near atmospheric pressure, circulating sodium coolant through its core (Figure 1-1), with heat transferred from the primary sodium coolant to an intermediate sodium loop. The reactor core design is based on decades of U.S.-based Sodium Fast Reactor (SFR) design and operational experience from the Experimental Breed Reactor II (EBR-II) and the Fast Flux Test Facility (FFTF) in addition to decades of technology development from the TerraPower TWR® technology and the Clinch River Breeder Reactor Plant (CRBRP) Program. The fuel for the Natrium core is sodium-bonded uranium-10 weight percent Zirconium (U-10Zr) that is encapsulated by HT9 ferritic-martensitic steel cladding. Historically, metallic fuel has been successfully demonstrated in the EBR-II reactor and metal fuel testing was conducted in the FFTF reactor core.

The RCC is designed to sustain a nuclear chain reaction for fission heat generation with a rated power of 840 MWt. Thermal power and reactivity are controlled by a combination of fuel management operations on a cycle-by-cycle basis and vertical positioning of the control rods throughout the operating cycle.

While operating, the reactor core generates and transfers heat to the primary sodium coolant in the Primary Heat Transport System (PHT). The Primary Sodium Pumps (PSPs) within the PHT supply sodium coolant flow to the core inlet plenum located at the bottom of the reactor core. The primary sodium coolant enters the core assemblies through their inlet nozzles, which interface with the receptacles located in the Reactor Enclosure System (RES) inlet plenum. The core assembly inlet nozzles, named Universal Stack Nozzles, contain orifice plates that control the primary sodium coolant flow through each core assembly to satisfy cooling requirements, which vary by assembly. A portion of the flow is designed to flow between the reactor core assemblies into the interwrapper region, also providing cooling of core assemblies stored in the In-Vessel Storage (IVS). As the sodium coolant exits each core assembly, it mixes in the hot pool volume within the Reactor Vessel (RV) and flows past the Upper Internal Structure (UIS). The sodium then flows into the Intermediate Heat Exchanger (IHX), where heat is transferred to the Intermediate Heat Transport System (IHT). The flow circuit is completed as the cooled sodium at the outlet of the IHXs enters the PSPs and is discharged back into the RES inlet plenum.

There are various core assembly types, positioned within the core based on the assembly function. Each assembly is removable to allow for relocation and replacement. The assembly types have similar shape and external configurations. Common features of core assemblies include a hexagonal cross section, an inlet nozzle, a duct tube, load pads, and a handling socket. The assembly duct and its load pad features, in conjunction with the core restraint system, are designed for proper core reactivity response while providing adequate inter-assembly clearances for fuel management during refueling operations. Assembly internal configurations are specific to the particular assembly type.

Additional description of these systems can be found in Section 7.1 of the PSAR and detail on the key design features, manufacturing parameters, and performance specifications for core assemblies can be found in NAT-2806 Rev 0, “TerraPower, LLC (TerraPower) Natrium Topical Report: Fuel and Control Assembly Qualification” [3].

Reactivity control is provided by nine primary control rod assemblies and four secondary control rod assemblies. These function in conjunction with the Control Rod Drive System (CRD) to position neutron absorber material to appropriately control and terminate the nuclear reaction. The Control Rod Driveline Mechanism (CRDM), control rod driveline, and control assembly are directly coupled during normal operation.

The control rods (located within 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 and maintain fuel within design limits. The control rods control core reactivity changes during normal reactor operation and accident conditions. The control rods provide scram insertion capability with sufficient reactivity worth to shut down the reactor and maintain it in a safe shutdown condition even if the highest worth rod is stuck in the withdrawn position.

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Figure 1-1: Coolant Flow Path through the RES and RCC

2 REACTOR DESIGN

This section provides an overview of the reactor core layout and modeling features of the core assemblies for performing nuclear and thermal hydraulic multi-physics simulations. The core model is preliminary and considers only the key core assemblies, such as fuel, control, reflectors, and shields for the purposes of characterizing system behavior for providing inputs to transient and safety analyses.

2.1 Core Model Description

Figure 2-1 provides an overview of the core assembly loading configuration for the first cycle of operation of the reactor core, whereas Table 2-1 provides the legend for each assembly.

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Figure 2-1: Core Assembly Loading for Cycle 1 (Assembly Pitch in Inches)

Table 2-1: Core Assembly Loading Legend for Cycle 1

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Figure 2-2: Illustration of Cascade Reloading Sequence

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Figure 2-3: Example Fuel Shuffling Cascade with Third Core Symmetry

While achieving an exact equilibrium state may not be possible through actual reactor operations, this evaluation approach provides key nuclear, thermal hydraulic, and fuel performance characteristics of the reactor core operations throughout its life. Figure 2-3 provides an example of the one-third core symmetric model (with a periodic transport boundary condition applied). simplify the fuel cycle evaluations. Fuel management and control rod replacement schedules are set to maintain this symmetry in the nuclear and thermal hydraulics calculations. For analyses that require asymmetries (e.g., individual control rod worth curves, reactivity coefficients, etc.) the reactor core model is expanded to a full core state. [[

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Figure 2-4: Core Inlet Nozzle Orifice Zone Map

There are multiple flow zones within the core, and these are mapped based on the combination of the inlet nozzles of the core assemblies and the receptacle inlets where they are placed. There are up to five orifice zones per nozzle, with the combination of nozzle and receptacle mapping to satisfy the core assembly positional flow requirements.

Table 2-2: Core Inlet Nozzle Orifice Map Legend

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Figure 2-5: Illustration of a Universal Stack Nozzle (Dimensions in Inches)

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2.2 Core Assembly Model Descriptions

There are five types of core assemblies modeled for nuclear and thermal hydraulics calculations. Figure 2-6 provides simplified vertical illustrations of the core assemblies with axial region labels and assignments of nuclear cross section labels used within the core design calculations. Note that fuel axial height increases with burnup, due to swelling, which is why it is “variable”. Example assembly designs are shown below in Figure 2-7 through Figure 2-11.

2.2.1 Fuel Assemblies

The Fuel Assembly (Figure 2-7) consists of a hexagonal duct structure with pins containing sodium-bonded fuel, cladding, and a helical wire-wrap that extends the length of the pin from the bottom to the top end plugs. The bottom of pinned region is attached to pin rails and pins contain an axial shield slug followed by fuel slugs and the fission gas plenum. The bottom and top of the assembly have inlet nozzle and handling socket hardware attachments.

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Figure 2-6: Vertical Arrangement of Modeled Regions within the Core Assemblies

Table 2-3: Heights and Regions Modeled of the Fuel Assembly

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Figure 2-7: Fuel Assembly Design

2.2.2 Control Assemblies

The Primary Control Rod Assembly (Figure 2-8 and Figure 2-9) consists of [[ ]](a)(4) The control pins are made of boron carbide neutron absorbing material, a helium gap, cladding, and a helical wire-wrap that extends the length of the pin from the bottom to the top end plugs.

The Primary Control Rod Assembly is inserted and withdrawn vertically from the top of the reactor core [[ ]](a)(4)

Similarly, the Secondary Control Rod Assembly (Figure 2-8 and Figure 2-9) consists of [[ ]](a)(4) The control pins are made of boron carbide neutron absorbing material, a helium gap, cladding, and a helical wire-wrap that extends the length of the pin from the bottom to the top end plugs.

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Figure 2-8: Control Rod Assembly Design

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Figure 2-9: Primary (Left) and Secondary (Right) Control Assembly Cross-Section

2.2.3 Reflector Assemblies

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Figure 2-10: Reflector Assembly Design

2.2.4 Shield Assemblies

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Figure 2-11: Shield Assembly Design

2.2.5 Materials Modeled

In modeling the assemblies described above, material data were developed and documented in a materials handbook for the following: U-10Zr, Boron Carbide pellets, SS316, HT9, Inconel 718, Sodium, and Helium.

Core design analysis utilizes this material property data to perform thermal expansion of components, calculate component number densities, and component temperature distributions for assessing system behavior against design limits. These properties are used when performing nuclear, thermal hydraulic, and mechanical evaluations using the codes discussed in Section 3.

2.3 Steady-State Conditions

An example evaluation of the steady-state conditions and characteristics of the Natrium reactor is presented in Section 8, with a summary of the key results provided in Table 2-4

Table 2-4: Preliminary Steady-State Results of the Natrium Reactor Core

Quantity	**Value**
Core Power[7]	840 MWt
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Inlet Temperature[2,7]	680°F (360°C)
Outlet Temperature[2,7]	950°F (510°C)

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3 CODES AND METHODS

3.1 Core Nuclear Design Codes and Methods

The core design codes and methods include a combination of industry codes together with methods developed internally for application to the Natrium reactor core. These are briefly described below and their use in application methodologies is described in Section 4.1.1.

When reviewing steady state neutronics phenomena information to develop the methods discussed below, there are a number of available historical SFR design and operating experiences for reference. As an example, the list of important phenomena was recently compiled in a report from Brookhaven [4]. These phenomena have been considered in the decades of Argonne National Laboratory (ANL) physics codes development, such as MC^2-3 (Section 3.1.3) and DIF3D (Section 3.1.2).

Another important source of method development comes from the best practices given in ANSI/ANS-19.3-2022 [5]. The computational sequence implemented by ANL physics codes is checked against the common practices for the Liquid Metal Reactor (LMR) core physics methods. The confirmation of agreement enhances the confidence in the current steady state nuclear methodology.

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3.1.2 DIF3D

DIF3D [10] is a fast reactor diffusion and transport analysis code developed by Argonne National Laboratory. This code is used to compute whole-core flux and power distributions (among other figures of merit, such as nuclide reaction rates, fast flux, etc.) for the core design, which in turn feed a substantial number of follow-on models and analyses.

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3.1.3 MC²-3

MC²-3 [11] is a multigroup cross section generation code for fast reactor applications. It is used to generate broad group cross sections, heating data, and delayed neutron data for various compositions in the reactor.

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3.1.4 PARTISN

PARTISN [12] was developed by Los Alamos National Laboratory and solves the multigroup discrete ordinates form of the Boltzmann transport equation in several different geometries. [[

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Figure 3-1: [[ ]](a)(4)

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3.1.5 MCNP

MCNP [14] was developed by Los Alamos National Laboratory and can perform a large variety of particle transport simulations, including fixed-source neutron, photon, and beta particle transport, and fission-source neutron transport calculations. [[

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3.2 Core Thermal Hydraulic Codes and Methods

Core thermal hydraulics uses a variety of tools with various levels of fidelity:

- One dimensional thermal hydraulic method to model core wide average temperatures, core flow distribution and pressure drop(Section 3.2.1).
- Subchannel code that provides more fidelity to the temperature inside each assembly as well as duct temperatures. This code can used for the entire core to provide detailed temperature distributions(Section 3.2.2).
- Computational Fluid Dynamics (CFD) to provide details of highly 3D phenomena like thermal mixing inside an assembly (Section 3.2.3).

3.2.1 Simplified Core Thermal Hydraulic Method

A set of simplified calculation methods was developed to perform steady-state thermal hydraulics calculations to aid in the design of the reactor core. Their intended use is to perform calculations to determine the steady-state temperature and pressure profiles in the core. [[

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3.2.1.1 Core Flow Distribution and Core Pressure Drop

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3.2.1.2 Assembly Pressure Drop

The assembly pressure drop is the sum of the orifice pressure drop, inlet/outlet form loss pressure drops, and pin assembly pressure drop. The following factors are computed and combined to determine overall pressure drop:

3.2.1.2.1 Assembly inlet and outlet losses

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3.2.1.3 Peak Fluid and Clad Temperature

Nascent [16] is an acronym for Natrium Simplified Coolant ENergy Transport and is a simple and fast-running tool for evaluating the peak coolant temperature in fuel assemblies under steady-state conditions. Nascent is used as an intermediate step in the calculation of the peak cladding temperature of the reactor fuel assemblies.

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3.2.1.4 Hot Channel Factors

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3.2.2 Mongoose++

Mongoose++ is a core thermal hydraulics subchannel analysis code developed internally by TerraPower for performing core thermal hydraulic analyses for single-phase sodium cooled reactors. The models implemented in the code are comprised of discretized mass, momentum, and energy conservation partial differential equations augmented with constitutive models.

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3.3 Core Mechanical Codes and Methods

3.3.1 Abaqus

Abaqus is a general-purpose commercial finite element analysis (FEA) software. Abaqus solves a range of equations for solid mechanics and heat transfer using a discretization of the problem domain which can represent arbitrarily complex geometries. It can provide solutions describing the behavior of systems and components under sophisticated boundary conditions and complicated material properties. Abaqus is capable of solving linear and nonlinear problems involving static, dynamic, viscoelastic, thermal, and fluid behavior response.

3.3.2 OXBOW

OXBOW is an internally developed code supporting core mechanical analysis. This software automates the pre- and post-processing of FEA models of a SFR core in order to evaluate the mechanical response of the core to a variety of load histories. OXBOW contains interfaces with commercially available finite element codes such as Abaqus and Ansys.

Together, Abaqus and OXBOW are used to model core assembly response and interactions as discussed in Section 4.2.

4 STEADY STATE CORE DESIGN ANALYSIS METHODOLOGY

This section describes the analysis methodologies associated with characterizing the reactor core during steady state operations and is applicable for rated and off-rated thermal and flow conditions. Section 4.1 describes the process for evaluating the representative characteristics of the core as a function of fuel management cycle and intermediate depletion evaluations. Following this, subsequent evaluations are performed to characterize the radial bowing reactivity feedback mechanism as described in Section 4.2, and Section 4.3 provides an approach for calculating neutron fluence.

4.1 Steady State Core Design

The results from steady state core design provide insight into the expected steady state nuclear, thermal hydraulic, and fuel performance characteristics of the Natrium reactor. It provides a basis for downstream core design calculations and supports design transient modeling and safety analysis.

Detailed nuclide inventories and material cross-sections are developed to support flux and power distributions throughout the core as described in Section 3.1, providing a basis for determining temperature and pressure profiles throughout the core as discussed in Section 3.2. This is generally performed at a simplified bundle subchannel level of calculations for evaluations of peak coolant temperatures, as is presented in the overall steady-state evaluation discussed below. Thermal hydraulic methods also support more detailed assessments o when other downstream analyses require specific information, such as a core restraint performance analyses that require detailed duct temperature evaluations (Section 4.2).

The overall steady state method is described below, while example results are provided in Section 8.

4.1.1 Neutronics

The overall solution strategy of a fast reactor physics calculation is analogous to the light water reactor (LWR) analysis methods in terms of the generation of multi-group cross sections and applying those to neutron transport solver. [[

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4.1.1.1 Nuclide Depletion

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4.1.1.2 Axial, Radial, and Local Power Distributions

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4.1.1.3 Reactivity Coefficients and Kinetics Parameters Calculation

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4.1.1.3.4 Reactivity Coefficient Calculations

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4.1.1.3.5 Number Density Perturbations

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4.1.1.3.6 Dimensional Perturbations

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4.1.1.3.7 Microscopic Cross Section Perturbations

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4.1.1.4 Control Rod Patterns and Rod Worth

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4.1.1.5 Reactivity Control System Analysis

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4.1.1.6 Decay Heat

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4.1.1.7 Stability

The methodology applied to analyze the stability of the Natrium reactor is described in the Stability Methodology Topical Report [2]. [[

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4.1.2 Thermal Hydraulics

Thermal hydraulic evaluations encompass a set of calculations that include coolant flow distribution, coolant temperatures, coolant pressures, and assembly component temperatures within the reactor core under normal operating conditions as described in the Simplified Core Thermal Hydraulic Method (Section 3.2.1). These are used in the design process to ensure that all core assemblies are adequately cooled.

4.1.2.1 Core Flow Distribution and Core Pressure Drop

The core flow distribution and core pressure drop are computed using the methods described in Section 3.2.1 with the inputs and assumptions described in this section.

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4.1.2.2 Assembly Pressure Drop

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4.1.2.3 Peak Fluid and Clad Temperature

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4.1.2.4 Hot Channel Factors

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Figure 4-1: Example relative temperature and uncertainty profiles with layered hot channel factor method through to fuel centerline.

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4.1.2.5 Assembly CFD Methodology

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4.1.3 Fuel Performance

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4.2 Core Restraint System Performance

Fast reactor cores are sensitive to fuel motion which occurs primarily due to core assembly bowing response to external loads (temperature, fluence, and contact). Core-wide assembly movements can cause reactivity changes which affect core power. The collection of load pads and their mechanical contacts restrain the core assemblies to control position and alignment .

As part of the core restraint system analysis, core assembly bowing deformations are used to predict the radial bowing reactivity feedback mechanism for a fresh core or a configuration that has undergone multiple cycles of operation. Between each cycle of operation, core assemblies are shuffled during refueling operations. After shuffling, core assemblies are arranged in a new configuration and the mechanical performance is evaluated based on the new locations, orientations, and inelastic history from previous cycles.

Figure 4-2 illustrates the mechanical bending/bowing induced by temperature gradients across core assemblies due to contact interactions at the ACLP and the top load pad. The following sub-sections describe the integrated approach for evaluating the reactivity feedback due to the core assembly deformations.

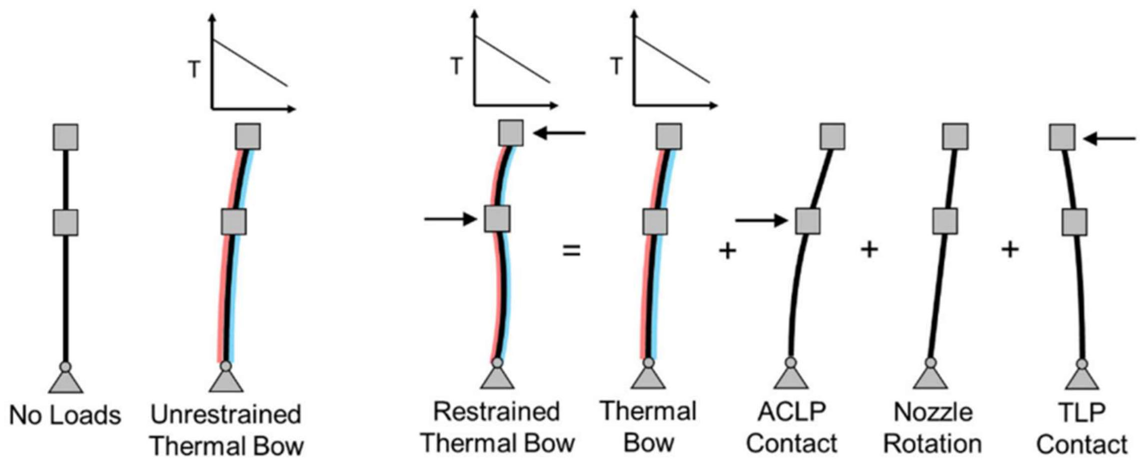

Figure 4-2: Example deformation shapes resulting from thermal bowing and assembly restraint

Due to the heterogenous geometry of the core, complex temperature/irradiation loading conditions, and unique material behaviors, it is unrealistic to obtain an analytical mathematical solution for such a complex system. Abaqus (Section 3.3.1) is used in conjunction with the internally-developed code OXBOW (Section 3.3.2) and utilizes the finite element method to solve these complex problems involving complicated geometries, loads, and materials.

The following subsections describe the calculations performed by various physics disciplines to contribute to the multi-physics core restraint analysis.

4.2.1 Thermal Hydraulics

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4.2.2 Mechanical

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4.2.3 Neutronics

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4.3 Neutron Fluence

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Figure 4-3: [[ ]]^(a)(4)

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5 METHODOLOGY VALIDATION

Methodology Verification and Validation (V&V) activities for the codes and methods discussed above are part of ongoing work and are planned to be complete prior to submittal of the operating license application. Assessment plans for validating the core nuclear and thermal hydraulic models of the steady state core design are provided in the following subsections.

5.1 Core Nuclear Design Method and Code Validation

Validation of nuclear design methodologies using the codes and methods in Section 3.1 is in development, with roadmaps in place to guide execution and future development. Validation will be achieved primarily through the usage of benchmark models based on historical reactor experiments, supplemented by the use of code-to-code comparisons with, for instance, MCNP and ORIGEN.

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Table 5-1: Overview of benchmark models for nuclear methods validation

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5.2 Thermal Hydraulic Method and Code Validation

Validation of the thermal hydraulic methodologies using the codes and methods in Section 3.2 is in development. Validation datasets include a mixture of legacy testing data, testing established by the thermal hydraulic test campaign for the Natrium project, and high-fidelity CFD. Thermal hydraulic validation assessments are preliminary at this phase of the design.

5.2.1 Legacy Tests

Data from a wide variety of legacy tests will be employed. [[

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5.2.2 Core Thermal Hydraulic Tests

Various testing has been identified supporting methodology V&V, hydraulic characterizations, and design development. [[

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5.2.3 High-fidelity CFD

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5.3 Core Mechanical Methods and Code Validation

Validation of core mechanical methodologies using the code and methods in Section 4.2 is in development. Benchmark assessments include a mixture of historical datasets (legacy data) and testing established by the Core Mechanical Test Campaign for the Natrium project. [[

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6 SUMMARY AND CONCLUSIONS

This report documents the core design and analysis methodologies which are used to perform nuclear design, thermal hydraulic, and mechanical calculations for the reactor, including reactivity coefficients, reactivity control system analysis, power distribution, and reactor core kinetics parameters. These methods apply to normal operation and may be used to inform license basis events for a Natrium reactor.

V&V of the core neutronics and thermal hydraulics codes is performed through comparisons with experimental results and to analyses from other codes. The uncertainty in the results from these codes is established based on industry experience and with a conservative bias due to the lack of operating experience with a Natrium reactor.

The completion of the V&V of the codes and methodology is part of ongoing work, and is planned to be complete prior to the submittal of an operating license application. In addition, the values of the uncertainties used in any applications will be documented as part of the safety analysis documents associated with the application.

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8 APPENDIX - EXAMPLE STEADY STATE CORE DESIGN RESULTS

8.1 Example Steady State Core Design Results

The following section provides example results from the methods discussed in the report above. These evaluations are performed based on a preliminary design for a Natrium pool-type, sodium-cooled, fast reactor. Table 2-4 provides a top level summary of these results, and Sections 2.1 and 2.2 provide a summary of the core geometry features that are used in evaluation.

8.1.1 Core Neutronics from Startup to Equilibrium

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Figure 8-1: k-effective as a Function of Time

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Figure 8-2: Maximum Core Assembly Burnup (%FIMA) vs. Time

8.1.2 Core Neutronic Performance at Selected State Points

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Figure 8-3: Total Assembly Power (MW) at BOL

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Figure 8-4: Maximum Fuel Assembly Burnup Distribution (%FIMA) at EOEC

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Figure 8-5: Maximum Assembly DPA at the EOEC

8.1.3 Nominal Pin Power Distributions

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Figure 8-6: Core Mid-Plane Pin Linear Power (W/cm) at BOL

8.1.4 Fuel Assembly Pin Power Peaking

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Table 8-1: Natrium Fuel Pin Power Peaking Factors

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Table 8-2: Fuel Assembly Maximum Pin Power and Burnup Peaking Factors

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8.1.5 Core Average Axial Power Profiles

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Figure 8-7: BOL Core Average Axial Power Shapes

8.1.6 Core Average Power Coefficient of Reactivity

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Table 8-3: Evaluated Core Average Full-Flow Power Coefficients of Reactivity

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8.1.7 Control Rod Evaluation Results

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Table 8-4: Third Core Critical Control Rod Withdrawn Elevation (Steps Withdrawn)

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8.1.8 Hot Full Power Individual Control Assembly Worth Curves

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Figure 8-8: HFP BOL All Rods Critical Individual Assembly Worth

8.1.9 Low Power Low Flow Critical Rod Height Evaluations

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Table 8-5: Low Power Low Flow Evaluations

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8.1.10 Reactivity Control System Analysis

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Table 8-6: Reactivity Control System Margin Results Summary

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Figure 8-9: ARI k_{eff} vs Isothermal Temperature at BOL

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Table 8-7: [[ ]]$^{(a)(4)}$

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Table 8-8: [[ ]](a)(4)

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8.1.12 Spatially Dependent Reactivity Coefficients

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8.1.12.1 Coolant Temperature Coefficient

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Figure 8-10: Assembly Total Coolant Temperature Coefficient at EOEC, m¢/K

8.1.12.2 Doppler Coefficients

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Figure 8-11: Assembly Total Doppler Coefficient at BOEC, m¢/K

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Table 8-9: [[ ]](a)(4)

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8.1.14 Core Decay Heat Results

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Figure 8-12: Core Decay Heat with Uncertainties and Biases [[

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8.1.15 Thermal Hydraulics

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Table 8-10: Flow Distribution by Zone at BOL

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Table 8-11: Pressure Drop by Zone at BOL

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Table 8-12: Peak Cladding Inner Diameter Temperature at BOL

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Figure 8-13: Subchannel Temperature Distribution at Outlet at BOL HFP Conditions

8.1.16 Hot Channel Factors

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Table 8-13: Hot Channel Factors for Fuel Assemblies

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Table 8-14: Hot Channel Factors for Primary Control Assemblies

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Table 8-15: Hot Channel Factors for Secondary Control Assemblies

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Table 8-16: Hot Channel Factors for Reflector Assemblies

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Table 8-17: Hot Channel Factors for Shield Assemblies

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