GE Hitachi Nuclear Energy

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BWRX-300 UK Generic Design Assessment (GDA) Chapter 4 - Reactor

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EXECUTIVE SUMMARY

The BWRX-300 reactor consists of the reactor pressure vessel, reactor internals, core, control rods, fine-motion control rod drives and in-core nuclear instrumentation. This chapter of the Preliminary Safety Report describes the components of the nuclear reactor core which includes the fuel assemblies, reactivity control systems and core monitoring system. Nuclear and thermal-hydraulic design aspects pertaining to the reactor core are also described. Design bases covering safety and performance aspects are specified for each area covered. The analytical methods and techniques used to evaluate the design are described. Finally, results from the evaluation of the design against the design bases are provided.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CAE	Claims, Arguments, Evidence
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDH	Control Rod Drive Housing
DEC	Design Extension Condition
DSA	Deterministic Safety Analysis
ECI	Export Controlled Information
ESBWR	Economic Simplified Boiling Water Reactor
FMCRD	Fine Motion Control Rod Drive
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
GNF	Global Nuclear Fuel
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
I&C	Instrumentation and Control
ICS	Isolation Condenser System
KKM	Kern Kraftwerk Müehleberg
LFWH	Loss of FeedWater Heating
LHGR	Linear Heat Generation Rate
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MFLPD	Maximum Fractional Limiting Power Density
MLHGR	Maximum Linear Heat Generation Rate
NBS	Nuclear Boiler System
OLMCPR	Operating Limit Minimum Critical Power Ratio
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PA	Postulated Accident
PCSR	Pre-Construction Safety Report

Acronym	Explanation
PIE	Postulated Initiating Event
PRNM	Power Range Neutron Monitoring
PSR	Preliminary Safety Report
P&ID	Piping and Instrumentation Diagram
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limit
SAP	Safety Assessment Principle
SCRRI	Selected Control Rod Rapid Insertion
SDC	Shutdown Cooling System
SDM	Shutdown Margin
SMR	Small Modular Reactor
TRAC	Transient Reactor Analysis Code
TRACG	GEH Proprietary Transient Reactor Analysis Code
UK	United Kingdom
U.S.	United States
USNRC	U.S. Nuclear Regulatory Commission
WRNM	Wide Range Nuclear Monitor

SYMBOLS AND DEFINITIONS

Symbol	Definition
°C	Degrees Centigrade
cm	Centimetre
Fe	Iron
K	Kelvin
k-effective	Neutron Multiplication Factor
kg	Kilogram
kgU	Kilogram of Uranium
kJ	Kilo-Joule
kPa	Kilo-Pascal
kW	Kilo-Watt
L	Litre
mm	Millimetre
MWth	Mega-Watt Thermal
Nb	Niobium
Sn	Tin
s	Second
UO_2	Uranium Oxide

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance
B	All	Editorial corrections, Proprietary Information and UK Export Control Information removal.

4. REACTOR

Preliminary Safety Report (PSR) Chapter 4 describes the components of the nuclear reactor core which includes the fuel assemblies, reactivity control systems and core monitoring system. Nuclear and thermal-hydraulic design aspects pertaining to the reactor core are also described. Design bases covering safety and performance aspects are specified for each area covered. The analytical methods and techniques used to evaluate the design are described. Results from the evaluation of the design against the design bases are also provided.

Claims and arguments relevant to Generic Design Assessment (GDA) step 2 objectives and scope are summarized in Appendix A, along with an As Low As Reasonably Practicable (ALARP) position. Appendix B provides a Forward Action Plan, which includes future work commitments and recommendations for future work where ‘gaps’ to GDA expectations have been identified.

4.1 Summary Description

The reactor assembly consists of the Reactor Pressure Vessel (RPV), pressure-containing appurtenances including Control Rod Drive (CRD) housings, in-core instrumentation housings and reactor internal components is shown in Figure 4-1. A summary of the important design and performance characteristics of the reactor and plant is given in PSR Ch. 1, Table 1-1.

A brief overview of the BWRX-300 reactor is provided in this section. The basis for reactor materials selection and overview of fabrication is described in PSR Ch. 5, Section 5.2. The reactor internal components and core support structures are described in PSR Ch. 5, Section 5.4. The RPV design is described in PSR Ch. 5, Section 5.4.

The fuel assembly (fuel bundle plus channel) is described in Section 4.2 including the design bases, analytical methods, and evaluation results. Aspects pertaining to the nuclear design of the core, including the reference equilibrium fuel cycle used for safety analyses, are described in Section 4.3. The thermal-hydraulic design basis requirements and associated methodologies are described Section 4.4, this section also discusses thermal-hydraulic stability. A description of the control rods and CRD system and associated requirements is provided in Section 4.5. The core monitoring function is described in Section 4.6.

4.1.1 Reactor Pressure Vessel

The BWRX-300 RPV is a vertical, cylindrical pressure vessel fabricated with forged rings with a removable top head by use of a head flange, seals, and bolting. The RPV also includes penetrations, nozzles, and reactor internals support.

The increased RPV height, relative to that for a typical forced circulation Boiling Water Reactor (BWR), is achieved by a “chimney” in the space that extends from the top of the core (top guide) to the entrance to the chimney head and steam separator assembly. The natural circulation flow resulting from the tall RPV results in adequate thermal margins during power operation and off- normal conditions as described in Section 4.3.

The RPV design and description are provided in PSR Ch. 5, Section 5.4.

4.1.2 Reactor Internal Components

The major reactor internal components consist of:

- Core components (control rods and nuclear instrumentation)
- Core support structures
- Steam dryer assembly
- Chimney
- Chimney head and steam separator assembly

The core components and selected core support structures are addressed within this Chapter, with the remainder of the reactor internal components being addressed in PSR Ch. 5.

Except for the Zircaloy in the reactor core, the reactor internals are stress corrosion-resistant stainless steels or other high alloy material. The fuel assemblies, control rods, chimney head and steam separator assembly, chimney assembly, steam dryers and in-core instrumentation assemblies are removable when the reactor vessel is opened for refueling or maintenance.

4.1.3 Reactor Core

The reactor core is made up of 240 fuel assemblies arranged to form an upright cylinder. Additionally, movable control rods are inserted or withdrawn for reactivity control. The fuel assemblies are comprised of hermetically sealed fuel rods in a square array along with upper and lower tie plates, water rods, fuel rod spacers, fuel channel and connecting components. The fuel assemblies are supported by the reactor internals. Each core cell consists of a control rod and four fuel assemblies that immediately surround it. Each core cell is associated with a four–lobed fuel support piece. Around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces. The four fuel assemblies are lowered into the core cell and, when seated, springs mounted at the tops of the channels force the channels into the corners of the cell, such that the sides of the channels contact the grid beams. Core lattice designations are based upon relative water gap size between adjacent fuel assemblies and dimensional characteristics of the basic fuel assembly and channel.

4.1.4 Fuel Assembly

The Global Nuclear Fuels GNF2 fuel assembly consists of 92 fuel rods and two large central water rods that occupy eight (8) fuel rod locations contained in a 10x10 array (i.e., 100 lattice locations). Fourteen fuel rod locations are occupied by part length fuel rods.

The fuel rod consists of uranium dioxide in the form of cylindrical pellets contained in Zircaloy tubing. The tubing is plugged, sealed, and welded at the ends to encapsulate the fuel. Fuel rods are pressurized internally with helium during fabrication to reduce clad creepdown and promote heat transfer.

The design of the fuel assembly is covered in Section 4.2.

4.1.5 Control Rod Assembly

The design of the control rods and the CRD mechanism is covered in Section 4.5.

4.1.6 Nuclear Instrumentation

The performance of the core is monitored by fixed neutron detectors located within the reactor core. The in-core nuclear instrumentation provides input to automatic reactor core control and protection functions. The BWRX-300 nuclear instrumentation consists of Power Range Neutron Monitoring (PRNM), Gamma Thermometers (GT), and Wide Range Neutron Monitor (WRNM) systems.

The PRNM provides neutron monitoring power signals to the SC1 Instrumentation and Control (I&C) protection systems. The PRNM also provides signals for post-accident monitoring purposes and, through isolated one-way optical data, links to the core monitoring three-dimensional power distribution program and to the control rod blocking systems.

GTs are in-core devices that convert local gamma flux to an electrical signal that supplies information required to calibrate the Local Power Range Monitors (LPRM) in the PRNM system.

The WRNM system is a redundant pair of industrial computers with a real time operating system that monitors the fixed neutron detectors in the core. [[

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The core monitoring function design is provided in Section 4.6 and PSR Ch. 7, Section 7.3.

4.1.7 Analysis Techniques

The analytical techniques employed in core design are comprised of the computer codes summarized in Table 4-1 and engineering design practices. The computer codes in Table 4-1 are further described in the relevant sections of this chapter and in PSR Ch. 3, Appendix G.

4.1.8 Comparison to Previous Boiling Water Reactor Designs

The BWRX-300 reactor core operates via natural circulation flow and is considered a composite design comprised of the most desirable features developed and previously applied to the BWR fleet. The key features are:

- There are 240 fuel assemblies arranged identically to the Kernkraftwerk Mühleberg (KKM) reactor core.
- The lattice type is the N-lattice that originated with the Advanced Boiling Water Reactor (ABWR). The N-lattice provides additional moderator volume in the intra-assembly bypass gap as compared to earlier lattice types.
- The core flow results from natural circulation and the nominal bundle flow during power operation is lower than forced circulation reactors.
- The core average power density is low compared to most forced circulation BWRs and approximately 20% lower than KKM (i.e., 870 MWth vs. 1097 MWth).
- The CRDs are fine motion that were developed for the ABWR.
- GTs are used in-lieu of the traversing in-core probe system for LPRM instrument calibration.
- The reference control rod type is the most modern commercially available control rod – the Ultra-HD.

While the exact configuration of the BWRX-300 reactor core is new, the configuration is similar to the BWR operating fleet, and the performance of all principal aspects have been proven in fleet application. Any differences from KKM are encompassed within the current approved nuclear methods.

4.2 Fuel Design

4.2.1 Fuel Assembly Description

The reference GNF2 fuel assembly components are described in detail in NEDC-34041P, "BWRX-300 GNF2 Fuel Assembly Mechanical Design Report," (Reference 4-1) and shown in Figure 4-2. The fuel assembly consists of a fuel bundle, a channel that surrounds the fuel bundle, and a channel fastener that attaches the channel to the bundle. The fuel rods and water rods are spaced and supported by upper and lower tie plates and intermediate spacers. The lower tie plate has the function of supporting the fuel assembly in the reactor. The upper tie plate has a handle for transferring the fuel bundle from one location to another.

The key attributes and materials used in the GNF2 fuel assemblies are listed in Table 4-2.

Water chemistry controls used to minimize adverse effects on fuel assembly materials are described in PSR Ch. 23 – Reactor Chemistry.

4.2.2 Fuel Bundle Description

The GNF2 fuel bundle is comprised of fuel rods that contain a tube for cladding that houses the UO_2 fuel pellets with some UO_2 pellets containing gadolinia. Each fuel rod is hermetically sealed with welded upper and lower end plugs. The cladding and end plug material is Zircaloy-2 with a zirconium liner for pellet-clad interaction resistance. All fuel rods are inerted with helium gas before sealing.

Normal Full-Length Rods

There are 78 normal full-length rod locations in the GNF2 fuel bundle and reside in holes in the upper and lower tie plates. An expansion spring is installed onto the upper end plug that interacts with the upper tie plate and exerts a downward force maintaining the axial position of the fuel rod while accommodating irradiation growth.

Tie Rods

The upper and lower tie plates are connected by eight fueled tie rods threaded into the lower tie plate and attached by nuts at the upper tie plate. The tie rods in each bundle have lower end plugs that thread into the lower tie plate and threaded upper end plugs that extend through the upper tie plate. Nuts and locking tab washers are installed on the upper end plug to secure the upper tie plate. The tie rods support the weight of the bundle during fuel handling operations when the assembly is lifted by the handle.

Gadolinia Rods

Gadolinia rods are essentially normal full-length fuel rods with normal fuel pellets except sections of the fuel column contain pellets with Gd_2O_3 homogeneously blended with the UO_2 powder. The resultant pellets function as a burnable neutron absorber controlling excess reactivity in the fresh fuel. [[

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Part-Length Rods

Partial length rods are affixed to the lower tie plate and terminate above a specified spacer. Partial Length Rods result in increased flow area in the upper regions of the fuel bundle for reduced pressure drop and improved stability compared to earlier fuel designs that did not include them. Partial Length Rods also improve nuclear efficiency by matching the axial hydrogen-to-fissile uranium (H/U) ratio in fuel with axially varying moderator density. Partial length fuel rods also reduce core reactivity in the cold condition and increase cold shutdown margins.

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Water Rods

The GNF2 fuel bundle is designed with two large circular water rods that are centrally located and occupy eight fuel rod lattice positions (each water rod occupies four fuel rod lattice positions). [[

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Spacers

[[ ]] The spacer spring force prevents fretting wear on the fuel rods due to fuel rod vibration. Flow diversion devices added to the top of the spacer improves liquid droplet deposition onto the surface of the fuel rods in the two-phase flow region.

The GNF2 bundle design has a non–uniform axial spacing of the eight spacers for improved droplet deposition in the annular flow regime while maintaining the rod positioning function. The spacer axial separation in the lower region of the bundle is established to prevent excessive fuel rod bow during operation. The spacers are positioned closer to each other in the upper region of the bundle for increased liquid droplet deposition in the annular flow region. [[

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Upper Tie-Plate

The upper tie-plate is a grid-like structure manufactured from type–304 stainless steel. It supports the weight of the fuel assembly and positions the upper ends of the full-length fuel rods laterally during operation and handling. [[

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Lower Tie-Plate including Debris Filter

The [[ ]] lower tie plate, in conjunction with the upper tie plate, supports the weight of the fuel assembly and positions the rod ends laterally during operation and handling. [[

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4.2.3 Channel and Channel Fastener Description

Channel

The channel is a square box-section 'sleeve' constructed from Zirconium alloy, which fits over the outside the fuel bundle. For the GNF2 fuel design to be utilized in the BWRX-300 the GNF proprietary zirconium-based alloy (NSF) is to be adopted, the naming is based on its constituent alloying elements i.e., Niobium (Nb), Tin (Sn) and Iron (Fe). The channel is open at the bottom and makes a sliding seal fit over the lower tie plate. At the top of the channel, two opposite corners have welded clips. These clips support the weight of the channel on the upper tie plate posts. One of the clips has a hole for attaching the channel fastener to the bundle. The channel design incorporates a uniform thickness bottom end. The remainder of the channel has corners, sidewalls and sidewall grooves at the control rod roller, and symmetric locations providing sufficient strength in the regions of highest stress while minimizing material that absorbs neutrons.

The channel performs the following functions:

- Forms the fuel bundle coolant flow path outer periphery
- Provides a surface for control rod guidance in the reactor core
- Provides structural lateral stiffness to the fuel bundle
- Controls, in conjunction with the lower tie plate, coolant bypass flow at the channel/lower tie plate interface
- Provides a heat sink during a LOCA
- Provides a stagnation envelope for in-core fuel sipping

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Fastener

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The channel fastener casting (or guard) fits over the top of the channel and bolts through the channel clip into the upper tie plate. The fastener guard serves as a reaction support for the leaf springs, provides a captive housing and lead-in for the fastener spring, and protects the springs from being overstressed. The fastener bolt attaches the channel to the bundle and remains captive in the casting, even if the fastener bolt were to fail.

4.2.4 Design Bases

Fuel Assembly

The fuel assembly is designed to:

- Support self-sustaining fission chain reaction, thereby producing energy in the form of heat
- Maintain its integrity to retain fission products generated in the fuel during normal operation and operational transients or during Anticipated Operational Occurrence (AOO) conditions

The fuel rod design considers all applicable effects such as fuel density changes, fission gas release, clad creep, and other physical properties which vary with burnup. The integrity of the fuel rods is achieved by designing to prevent excessive fuel temperatures, internal gas pressure due to fission gas releases, cladding stresses, strains, and strain fatigue. The detailed fuel design also establishes such parameters as pellet size and density, clad/pellet diametral gap, gas plenum size, and helium pre- pressurization level.

The fuel assembly structure integrity is assured by setting limits on stresses and deformations due to various loads and by preventing the assembly structure from interfering with the functioning of other components. The fuel assembly is designed to withstand the following loads:

- Normal and abnormal loads occurring in startup testing, normal operation, AOOs
- Abnormal loads occurring in infrequent events and accidents

4.2.5 Design Evaluation

The GNF2 fuel design is the result of over 50 years of BWR fuel design, fabrication, and operational experience. The BWR fuel design process, comprised of engineering methods, analyses and test, is mature and generically applicable to the BWRX-300. The methods applied in the design of BWR fuel are documented in various Licencing Topical Reports that were submitted to the United States Nuclear Regulatory Commission (USNRC) as needed. In addition, GNF have previously developed a fuel licensing framework with the USNRC, and other regulatory authorities, called GESTAR NEDE-24011-P-A-31, “General Electric Standard Application for Reactor Fuel (GESTAR II),” (Reference 4-3). The GESTAR licensing framework defines the generic requirements and approved methods for designing BWR fuel.

Fuel Thermal-Mechanical Analytical Method

Most of the fuel rod thermal mechanical design analyses are performed using the PRIME fuel rod thermal-mechanical methodology. The method and qualification of the PRIME methodology are described in NEDC-33256P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 1 – Technical Bases,” (Reference 4-4) and NEDC-33257P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 2 – Qualification,”(Reference 4-5), respectively. The application of the PRIME methodology for the analysis of fuel rod thermal-mechanical performance is described in NEDC-33257P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 3 – Application Methodology,” (Reference 4-6) (steady state conditions) and NEDC-33840P, “The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance,” (Reference 4-7) (transient conditions). This methodology is applicable to the analysis of the fuel rod response for all AOO events and has been qualified for performing fuel analysis for the BWRX-300. PRIME analyses are performed for the following conditions:

- For each analysis, fuel rod input parameters are based on either the most unfavorable manufacturing tolerances (i.e., ‘worst tolerance’ analyses) or statistical distributions of the input values. Calculations are performed providing either a ‘worst tolerance’ or statistically bounding tolerance limit for the resulting output parameter(s)

- Operating conditions are postulated that cover the conditions anticipated during normal operating conditions and AOOs

Fuel rod design evaluations establish an upper bound power history envelope for the different fuel rod types. These power histories are used for fuel rod thermal mechanical design analyses evaluating the fuel rod design features and demonstrating conformance to the design criteria. These power histories are applied as a design constraint in the development of any BWR core design, including the reference BWRX-300 core described in Section 4.3.

The PRIME analysis assumes limiting operation during the fuel rod operating lifetime. For most analyses, the fuel rod (axial) node with the highest power operates on the limiting power-exposure envelope.

4.2.6 Worst Tolerance Evaluation

The analyses apply worst tolerance assumptions to the cladding circumferential strain during an AOO. In this case, the important PRIME inputs are all biased to the fabrication tolerance extreme in the direction that produces the most limiting circumferential strain. The biases are discussed in detail in NEDC-33257P-A (Reference 4-6).

4.2.7 Statistical Evaluation

PRIME analyses are performed using standard error propagation statistical methods.

4.2.8 Fuel Lift and Seismic and Dynamic Load Evaluation

The fuel lift and seismic and dynamic load analyses are completed prior to fuel release for transportation. The fuel lift seismic and dynamic load analysis evaluates whether these loads are sufficient to unseat the fuel assembly from the lower tie plate, and if so, the maximum vertical lift distance and maximum fuel dynamic acceleration resulting from reseating. The acceptance criteria for this analysis are:

- Vertical fuel lift is less than the engagement between the fuel support and the lower tie plate
- Peak horizontal and vertical accelerations are less than the fuel demonstrated acceleration capabilities

4.2.9 Cladding Strain Evaluation

The cladding strain analysis is performed using the PRIME code and the worst tolerance methodology discussed previously. For each fuel rod type the cladding strain is calculated at various exposure points where transient overpower is assumed relative to the limiting power history. The magnitude of the overpower event is increased until the cladding strain approaches but does not exceed the limits described in NEDC-33257P-A (Reference 4-6) at the most limiting exposure to establish the mechanical overpower. Mechanical overpowers may also be defined as a function of exposure, ensuring that cladding strain does not exceed the limits described in NEDC-33257P-A (Reference 4-6) at any exposure point.

4.2.10 Fuel Rod Internal Pressure Evaluation

The fuel rod internal pressure analysis is performed using the PRIME code and the statistical methodology discussed previously. The fuel rod internal pressure nominal value and standard deviation are determined at various fuel rod exposure points. At each of these exposure points, the nominal values and standard deviation of the fuel rod internal pressure that would cause the cladding to creep outward at a rate equal to the fuel pellet irradiation swelling rate (the critical pressure) are calculated. A design ratio is defined from the rod internal pressure and the critical pressure with their standard deviations. This design ratio limit of no more than 1.0 at a 95% confidence level, ensures that the fuel rod cladding does not creep out at a rate greater than the fuel pellet irradiation swelling rate.

4.2.11 Fuel Pellet Temperature Evaluation

The fuel pellet temperature analysis is performed statistically using the PRIME code. The fuel temperature analysis also reflects continuous operation. The maximum permissible overpower is determined considering the difference between the fuel melting temperature and the fuel pellet centerline temperature at the overpower condition. A mean value and a standard deviation for the difference between the fuel melting temperature and the fuel pellet centerline temperature are determined. The percentile of this difference is required to be greater than or equal to zero to ensure that fuel pellet centerline melting does not occur in the event of a specified overpower during plant operation.

4.2.12 Cladding Fatigue Evaluation

The cladding fatigue analysis is performed statistically using the PRIME code. Variations in power, coolant pressure and coolant temperature are superimposed on the limiting power history for calculating the cladding fatigue.

Cladding strain cycles are analyzed in NEDC-34042P, “BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report,” (Reference 4-8) using the fuel duty cycles shown in NEDC-33257P-A (Reference 4-6). The duty cycles represent conservative assumptions regarding power changes anticipated during normal reactor operation and AOOs, planned surveillance testing, normal control rod maneuvers, shutdowns and load following. Fatigue analyses explicitly account for daily load following using the BWRX-300 plant duty cycles specification. The analyses of GNF2 fuel show that the cladding fatigue capability accommodates the expected fatigue duty for the projected life and operation of the fuel with sufficient margin (Reference 4-8).

4.2.13 Cladding Creep Collapse Evaluation

Cladding creep collapse occurs when excessive in-reactor fuel pellet densification causes fuel column axial gaps. The cladding creep collapse analysis consists of a detailed finite element mechanics analysis of the cladding. This evaluation is described in NEDC-33139P-A, “Cladding Creep Collapse,” (Reference 4-9).

4.2.14 Fuel Rod Stress Evaluation

The fuel rod stress analysis is performed using the Monte Carlo statistical methodology and addresses local fuel rod stress concerns, such as the stresses at spacer contact points, that are not directly addressed by the PRIME code. Results from PRIME analyses are used to generate inputs for the stress analysis. The cladding stress analysis is described in, NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” (Reference 4-10).

4.2.15 Thermal and Mechanical Overpowers Evaluation

Analyses are performed to establish the values of the maximum overpower magnitudes that do not result in violation of the cladding circumferential strain criterion or the incipient fuel centerline melting criterion. As part of the core design and AOO analysis, the calculated core mechanical and thermal overpowers defined in NEDC-33457P-A (Reference 4-6) and NEDC-33840P (Reference 4-7) are compared with the thermal overpower and mechanical overpower criteria and confirm conformance to these criteria. PRIME transient analyses performed per NEDC-33840P (Reference 4-7) may also use worst-tolerance and statistical assumptions, consistent with those used in determination of thermal overpower and mechanical overpower criteria. This method confirms compliance by comparing worst-tolerance strain directly to strain limits and lower 95% melt margin criterion.

4.2.16 Fretting Wear Evaluation

Mechanical testing is performed to ensure that loss of fuel rod and assembly component in order to maintain the mechanical integrity due to fretting wear when operating in an environment free of foreign material.

The GNF2 fuel assembly was tested to assure that the design features do not result in a significant increase in flow induced vibration response and thereby do not increase the potential for fretting. The method used to demonstrate the adequacy of the fuel assembly from a flow induced vibration perspective was to compare the vibration response of the GNF2 design with the GE14 design during flow induced vibration tests. The response comparison was based on accelerometer data from various locations in the fuel assemblies. The GE14 fuel assembly's performance is considered acceptable based upon its reliable performance in reactor operation.

4.2.17 Water Rods Evaluation

Analyses are performed to determine component stresses at the bounding load conditions and compared to applicable criteria, such as yield and ultimate stresses. The load conditions consider shipping and handling loads, seismically induced bending moment, and the pressure differential across the water rod. The design is also evaluated using finite element analysis to determine the critical buckling load to ensure axial loads resulting from differential growth of water rods and other fuel assembly components are adequate.

4.2.18 Tie Plates Evaluation

Tie plate adequacy is demonstrated by detailed finite element analysis and mechanical testing for bounding fuel handling and seismic load conditions.

4.2.19 Spacers Evaluation

Cyclic testing for seismic loading demonstrates that the GNF2 spacer stresses and strains do not exceed failure values and that the fatigue capability is not exceeded. The results of flow induced vibration testing for GNF2 are summarized in NEDC-33270P (Reference 4-10).

4.2.20 Channel Evaluation

Channel adequacy relative to applicable design criteria is confirmed by performing the following evaluations:

- Calculating elastic stress and deflection caused by channel wall pressure difference
- Calculating thermal stresses due to the various temperature gradients that the channel is subjected to during normal operation and handling
- Calculating fatigue and stress rupture considering the combined effect of pressure temperature cycling and hold time
- Calculating elastic-plastic and creep of channel wall permanent deflection
- Calculating channel stress due to control rod contact
- Analyzing channel/lower tie plate differential thermal expansion

The GNF2 Fuel Assembly Mechanical Design Report for BWRX-300 (Reference 4-1) provides additional discussion addressing the design features that preclude excessive channel bowing from preventing control rod insertion.

4.3 Nuclear Design

4.3.1 Description

The core of the BWRX-300 is light-water moderated and fueled with low-enriched uranium dioxide fuel assemblies. The use of light water as a moderator produces a neutron energy spectrum where fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The negative void reactivity feedback effect is an inherent safety feature of BWRs. Any system change that increases reactor power, either locally or core-wide, produces additional steam voids and reduces power.

The reactor core is arranged as an upright cylinder containing 240 fuel assemblies located within the core shroud. The coolant flows upward through the core. The reactor core includes fuel assemblies, control rods, and nuclear instrumentation. The arrangement of fuel assemblies and control rods along with in-core instrumentation is the same as the forced flow Kernkraftwerk Mühleberg Nuclear Power Plant (KKM) BWR in Switzerland and is displayed in Figure 4-4.

The core nuclear design of any BWR core for an operating cycle is comprised of the following elements:

- Core arrangement, lattice type and fuel product line that combine to establish the detailed geometry of the reactor core
- The energy utilization plan that defines the energy requirements
- Core coolant hydraulics (e.g., core flow, pressure, and inlet temperature)
- Nuclear design (i.e., enrichment and burnable poison distribution) of the fresh fuel
- Core loading pattern of fresh and irradiated fuel
- Planned control rod patterns during power operation

An equilibrium cycle is a reactor state in which the same fresh fuel assemblies are loaded into the same locations, the irradiated fuel is shuffled to the same locations and the cycle is depleted in the same way until an equilibrium state is achieved producing essentially identical results (e.g., energy generated, power distributions, reactivity/thermal margins, etc.) cycle after cycle. An equilibrium cycle is the best representation of the core over the life of the reactor. A reference equilibrium core as shown in Figure 4-5 has been developed for safety analysis to demonstrate that the BWRX-300 conforms to all regulatory requirements with high confidence. The reference BWRX- 300 equilibrium cycle has been selected to be an annual cycle (i.e., 12-month refueling interval) with high, albeit normal, discharge exposure. Alternate refueling intervals (e.g., 18, or 24 months) may be applied to the BWRX-300 which can be demonstrated to conform to all applicable safety and performance requirements.

The reference equilibrium cycle is loaded with multiple fresh nuclear fuel bundle types of various enrichments and gadolinia burnable poison designs that satisfy a multitude of objectives, including to optimize the core burnup while maintaining core performance. The core loading pattern, operating control rod patterns, and core performance results are described in NEDC-34160P, “Reactor Core Nuclear Design Report,” (Reference 4-11) and the nuclear design of the fresh fuel bundles is provided in NEDC-34045P, "BWRX-300 GNF2 Fuel Bundle Information Report for Equilibrium 12-Month Cycle,” (Reference 4-12). Selected nuclear design information and core performance results from the reference 12-month equilibrium cycle are provided in Figure 4-5 to Figure 4-10.

Refueling Interval

BWRs operate on distinct refueling intervals. At the end of a normal operating cycle, after shutdown, the reactor is disassembled, then generally the lowest reactivity fuel is discharged and new reload fuel is inserted and the fuel is shuffled into the core configuration for the upcoming cycle. The reactor is then reassembled, and the startup of the next cycle can commence. This activity, from shutdown of any operating cycle to startup of the next, is termed the refueling outage. Inspections and maintenance are also typically performed during the refueling outage on intervals specific to each piece of equipment or component.

For BWRs, there is not a specific limit on the length of an operating cycle, or planned refueling interval; however, requirements governing inspection and surveillance frequency must be satisfied and the safety analyses that constitute the safety basis must encompass the planned operating cycle length. Cycle specific operating limits are established to span the planned operating cycle length and provision is made for cycle extension (e.g., power coast down) as specified by the reactor owner. Any planned operating cycle length must:

- Be aligned with the inspection frequency set out in the safety case for the reactor internals and equipment
- Reside within evaluated space (i.e., the technical inputs to the safety analyses must envelope the planned operating cycle length)

The cycle specific operating limits are derived from AOO analysis of the actual core design. Any refueling interval that results in acceptable thermal operating limits that conform to the Specified Acceptable Fuel Design Limits (SAFDL) can be supported.

BWRs operate on approximately 12-, 18-, or 24-month refueling intervals; however, operating cycles longer than 24 months have been conducted as well as intermediate lengths. The reference BWRX-300 equilibrium core design to support safety and performance evaluations was established to be a 12-month cycle because it illustrates the highest degree of operational flexibility associated with excess thermal margins.

4.3.2 Design Bases

The design bases require the plant to operate while meeting all safety limits:

- The reactivity bases ensure that uncontrolled positive reactivity excursions of the core are prevented
- The overpower bases ensure the core to operate within fuel integrity limits

4.3.3 Reactivity Feedback Design Bases

Reactivity coefficients representing the differential changes in reactivity produced by differential changes in core conditions are used for calculating stability parameters and evaluating the response of the core to external disturbances. The base initial condition of the system and the Postulated Initiating Event (PIE) determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest are the Doppler, moderator temperature, and the moderator void reactivity coefficient. Also associated with the BWR is a power reactivity coefficient. The power coefficient is a combination of the Doppler and moderator void reactivity coefficients in the power operating range and is not explicitly evaluated.

The fuel reactivity acceptance criteria are established in NEDE-24011-P-A (Reference 4-3) and each of the following fuel parameters must be negative throughout the life of the core:

- Doppler reactivity coefficient for all operating conditions
- Core moderator void reactivity coefficient resulting from boiling in the active flow channels for any operating conditions

- Moderator temperature coefficient for temperatures equal to or greater than hot standby
- Power coefficient, as determined by calculating the reactivity change resulting from an incremental power change from a steady–state base power level for all operating power levels above hot standby
- Net prompt reactivity feedback originating from prompt heating of the moderator and fuel for a super prompt critical reactivity insertion accident (e.g., control rod drop accident)

The Doppler coefficient, the moderator void coefficient and the moderator temperature coefficient of reactivity are negative for power operating conditions, thereby assuring negative reactivity feedback characteristics.

4.3.4 Shutdown Reactivity Margin Design Bases

The core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn or rod pair associated with the common Hydraulic Control Unit (HCU) postulated stuck in the full out position, and all other rods fully inserted. The Shutdown Margin (SDM) is determined by using the BWR simulator code (see Section 4.3) to calculate the core multiplication at selected exposure points with the strongest rod fully withdrawn. The SDM is calculated based on the carryover of the minimum expected exposure at the end of the previous cycle. The core is assumed to be in the cold, xenon–free condition in order to ensure that the calculated values are conservative.

4.3.5 Overpower Design Bases

It is required that the core operates within an absolute power and power distribution envelope to ensure fuel integrity is maintained. This is controlled through two nuclear design basis parameters: the Maximum Linear Heat Generation Rate (MLHGR) Limit and Minimum Critical Power Ratio (MCPR). The MCPR and Linear Heat Generation Rate (LHGR) limit are determined with 95% confidence that the fuel does not exceed Operational Limits and Conditions during AOOs. These constraints must then be met under normal operating conditions. Explicit Maximum LHGR Limit and MCPR parameter definitions are provided as follows.

Maximum Linear Heat Generation Rate

The LHGR limit is the maximum allowable linear heat generation for each fuel rod in the bundle. The LHGR operating limit is bundle type- dependent and is a function of gadolinia content and exposure. The LHGR is monitored, and the fuel is not operated at MLHGR values greater than those found acceptable by the safety analysis under normal operating conditions. Under AOO conditions, including the maximum overpower condition, the calculated overpower is confirmed to neither cause fuel melting nor exceed the stress and strain limits as discussed in Section 4.2.5.

Minimum Critical Power Ratio

MCPR is the minimum Critical Power Ratio (CPR) allowed for a given bundle type to avoid boiling transition. CPR is a function of several important parameters: bundle power, bundle flow, rod power peaking distribution, and bundle mechanical design. The plant Operating Limit Minimum Critical Power Ratio (OLMCPR) is established by considering the limiting AOOs for each operating cycle. The OLMCPR is determined to avoid boiling transition for 99.9% of the rods during the limiting analyzed AOO transient discussed in PSR Ch. 15, Section 15.5.

4.3.6 Design Evaluation

4.3.7 Core Nuclear Analytical Methods Evaluation

The analytical methods used in the design and analysis of the BWRX-300 core during all states of normal operation are summarized below and described in detail in, NEDC-34039P, "BWRX-300 GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology," (Reference 4-13).

The principal tools used in the steady-state nuclear core analysis are the three-dimensional (3D) BWR Core Simulator PANAC11 and the two-dimensional lattice physics code TGBLA06. The BWR Core Simulator is a coupled nuclear-thermal-hydraulic computer program representing the BWR core exclusive of the external flow loop.

Natural circulation flow is determined by the Transient Reactor Analysis Code General Electric (TRACG) described in NEDE-32176-P, "TRACG Model Description," (Reference 4-14). The associated core flow is used as an input to the core simulator.

The simulator computes core reactivity, power distributions, exposure, and reactor thermal- hydraulic characteristics, with spatially varying voids, control rods, and burnable poisons as described in the nuclear libraries and other constitutive state variables. The simulator is used to calculate reactivity variations through the cycle, shutdown margins and compliance with thermal limits (i.e., LHGR Limit and MCPR).

PANAC11 and the associated nuclear libraries produced by TGBLA06 have undergone extensive validation by comparing calculated results with alternate methods, end-of-cycle gamma scan data, and operating reactor data. PANAC11 is a well-established method used for production core design, licensing analysis, and core exposure tracking for the BWR fleet. The exposure tracking process provides the opportunity for continuous comparison and validation of the nuclear methods against operating data. Validation of the adequacy of PANAC11 is demonstrated in NEDC-34039P (Reference 4-13).

The lattice physics and 3D core simulator are used in computing the change in neutron multiplication (i.e., reactivity inserted) caused by a change in state (e.g., change in fuel temperature, in-channel void fraction, etc.) when determining the reactivity coefficients.

4.3.8 Core Reactivity Characteristics Evaluation

Reactivity Coefficients

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in calculating stability and evaluating the response of the core to external disturbances. The base initial condition of the system and the PIE determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest, relative to BWR systems, are discussed here individually.

The reactivity coefficients evaluation is performed as part of new fuel design development assuring consistency with safety objectives. The GNF2 results are documented in NEDC-33270P (Reference 4-10), which concludes that all the criteria defined in GESTAR II have been met for the GNF2 fuel design.

Reactivity Variation

The excess reactivity needed to deliver the target cycle energy while maintaining rated thermal power is controlled by the control rod system supplemented by fuel rods containing a burnable absorber. When applied to any specific fuel cycle, these integral fuel burnable absorber rods are used to provide partial control of the excess reactivity available during power operation. The burnable absorber lowers the reactivity of fresh fuel and is designed to be largely depleted by the end of the first cycle of operation. Control rods are used during the cycle to compensate for the remaining hot excess reactivity and reactivity changes due to burnup. Control rods may also be used to control the power distribution. The burnable absorber design is established such that the remaining hot excess reactivity is consistent with target control patterns during steady-state power generation. The control rod pattern and insertion program associated with the reference equilibrium cycle are presented in Figure 4-6 and Figure 4-7, respectively.

Control Reactivity

Neutron absorbing control rods are the primary means to control reactivity in transient and accident analyses. During events that result in relatively fast positive reactivity feedback, control rods are inserted rapidly using stored hydraulic energy. This is referred to as a scram. During events that result in relatively slow positive reactivity feedback and do not require a reactor “scram”, the Fine Motion Control Rod Drives (FMCRDs) that are operated using electric motors can be used for slower control rod insertion, (see Section 4.5.3). Reactivity feedback mechanism is key for some PIE groups (see PSR Ch. 15, Table 15.2-1) in which rods are withdrawn in error. Control reactivity is modeled using 3D kinetics coupled with the thermal hydraulic response in TRACG.

Doppler Reactivity

Doppler reactivity is a reactivity feedback mechanism in BWR transient, and accident analyses associated with changes in fuel temperature. Doppler reactivity is negative with an increase in fuel temperature and becomes more important as the fuel temperature continues to increase. In Deterministic Safety Analysis (DSA) where reactivity feedback is important, doppler reactivity is modeled using 3D kinetics coupled with the thermal hydraulic response in TRACG.

Void Reactivity

Void reactivity is an important reactivity feedback mechanism in BWR transient and accident analyses. The void reactivity feedback is always negative and is typically stronger (more negative void coefficient) at the end of an operating cycle. It is also stronger for reload cores versus an initial reactor core that includes only fresh fuel. This reactivity feedback mechanism is the dominant feedback for some PIE groups. PSR Ch. 15, Table 15.2-1 focuses on events initiated from conditions of normal power operation. In the DSA, where it is important, void reactivity is modeled using 3D kinetics coupled with the thermal hydraulic response analyzed using TRACG, the primary DSA computer code (see PSR Ch. 5, Section 15.5).

Moderator Temperature Reactivity

The moderator temperature coefficient of reactivity is defined as the change in reactivity produced by a unit change in moderator temperature. The value of this coefficient is important during the startup of a BWR. During power operation, the coefficient is not important, because the moderator is boiling, and primarily remains at the saturation temperature corresponding to the operating pressure.

Xenon Reactivity

Xenon reactivity feedback is not typically accounted for during events in DSA because the rate of change of reactivity is slow. The effects of xenon are accounted for in analyses of shutdown margin. Xenon reactivity impacts are also considered during core design and monitoring.

4.3.9 Shutdown Margin Evaluation

The core must be capable of being made subcritical with margin in the most reactive condition throughout an operating cycle with the most reactive coupled control rod pair in their full out position and all other control rods fully inserted. This calculation is performed at cold temperatures, which are between 20°C and 286°C. [[

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4.3.10 Thermal Limits Evaluation

The margins to thermal limits for the reference equilibrium cycle are presented in Figure 4-9 for Minimum Critical Power Ratio and Figure 4-10 for Maximum Fractional Limiting Power Density (MFLPD) and documented in NEDC-34160P (Reference 4-11).

4.3.11 Core Stability Evaluation

Xenon Stability

BWRs are not susceptible to xenon oscillations. The xenon stability evaluation has been demonstrated by:

- No observed xenon instabilities in operating BWRs
- Special tests conducted on operating BWRs forcing the reactor into xenon instability demonstrate that xenon transients are highly damped by the large negative moderator void feedback
- Simulation calculations

All these indicators demonstrate that xenon transients are highly damped in a BWR due to the large negative moderator void feedback. Moreover, the BWRX-300 reactor core, fueled with GNF2, is evaluated to be less susceptible to xenon oscillations as compared to KKM (that never experienced an oscillation mode) as the void fraction is expected to be higher and the corresponding void reactivity coefficient more negative.

Thermal-Hydraulic Stability

The most limiting stability condition in the BWRX-300 normal operating region is at rated power/flow condition. The BWRX-300 core remains stable throughout the entire operating domain. Refer to Section 4.4.23 for a discussion of thermal-hydraulic stability.

4.4 Thermohydraulic Design

4.4.1 Description

4.4.2 Reactor Coolant System Description

The Reactor Coolant System is described in PSR Ch. 5. The BWRX-300 reactor coolant system is shown in PSR Ch. 5, Figure 5-1. The BWRX-300 thermal hydraulic design is similar to operating BWRs except that it does not require recirculation pumps or associated coolant piping. Circulation of the reactor coolant through the BWRX-300 core is accomplished via natural circulation. Elevated natural circulation flow is promoted mainly by the addition of a tall chimney between the top of the core at the top guide plate level and the steam separator assembly. The natural circulation flow rate depends on the difference in water density between the core/chimney region and the downcomer region. The core flow varies according to the core power level because fluid density in the core/chimney region changes with the power level. Therefore, a core power-flow map reduces to a single line and there is no active control of the core flow at any given power level, as shown in Figure 4-11.

The values for a number of reactor thermal-hydraulic parameters of interest for the BWRX-300 compared to earlier BWR designs, including the ABWR, are given in Table 4-3.

4.4.3 Core Hydraulics Description

Control of fuel bundle flow is achieved through the use of orificed fuel supports (see Figure 4-12) which are distributed through the core as shown in Figure 4-13.

Accurate prediction of bundle flow and power distributions is important in the calculation of margin to the thermal limits of each fuel bundle. Pressure drop characteristics are included in plant cycle specific analyses for the calculation of the Operating Limit MCPR.

Because of the channeled configuration of BWR fuel assemblies, there is no assembly-to- assembly cross flow inside the core. The only issue of hydraulic compatibility of various bundle types in a core is the bundle inlet flow rate variation and its impact on margin to thermal limits (i.e., MCPR or MLHGR). The coupled thermal–hydraulic–nuclear analyses are performed each cycle to determine fuel bundle flow and power distribution. The analyses use the various bundle pressure loss coefficients to determine the flow distribution required to maintain total core pressure drop boundary conditions applied to all fuel bundles. The margin to thermal limits of each fuel bundle is determined using this consistent set of calculated bundle flow and power.

The flow distribution to the fuel assemblies and bypass flow paths is calculated using various pressure drop models that include friction loss coefficients, local loss coefficients, two-phase multipliers, and void-quality correlations. These models are developed from pressure drop data with a best-fit basis. Pressure drop measurements made in operating reactors confirm that the total measured and calculated core pressure drops agree. This information is collected normally as part of core management activities and its purpose is to identify anomalous behavior. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass flow paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass flow paths. The remainder passes through the orifice in the fuel support casting and/or peripheral fuel support (experiencing a pressure loss) where some of the flow exits through the fit–up between the fuel support and lower tie plate and through the lower tie plate holes into the bypass flow region. All fuel bundles have lower tie plate holes. Most of the flow continues through the lower tie plate (experiencing a pressure loss) where some flow exits through the flow path defined by the fuel channel and lower tie plate into the bypass region.

The unique GNF2 fuel assembly hydraulic characteristics include the inlet orifice, lower tie plate, upper tie plate, spacers, water rod, and various leakage paths. The hydraulic characteristics of these components flow paths have been developed and confirmed by test comparisons. These unique GNF2 hydraulic characteristics are used in all analysis models and methods where the fuel assembly hydraulics are needed.

The analytical methods used in the analysis of the BWRX-300 reactor are GNF standard codes in use throughout the industry and licenced in other jurisdictions.

Flow pressure drop characteristics are included in plant cycle specific analyses for the calculation of the Operating Limit MCPR.

4.4.4 Core Response Under Transient Conditions Description

Core and fuel response to postulated AOOs and accident events is explicitly modeled via the TRACG computer code employing three-dimensional kinetics that is consistent with the BWR Core Simulator PANAC11.

4.4.5 Reactivity Coefficients including Power Coefficient of Reactivity Description

The principal reactivity coefficient for the BWRX-300, and any BWR, is the void coefficient, which is the dominant constituent of the power coefficient of reactivity and required to be negative under all reactor states.

Power control to ensure compliance with LHGR limits, including aspects of loss of reactivity control, is based on the use of cruciform shaped control rods (which are sometimes referred to as rods), like other BWR types, for reactor thermal power, power distribution, and control of the BWRX-300 core. Fast acting shutdown capability is provided using stored hydraulic energy HCUs as a diverse motive force relative to the FMCRDs, as described in Section 4.5. The Core Monitoring function (PSR Ch. 7, Section 7.3) in concert with reactor operators ensure compliance with requirements implemented by the thermal limits MLHGR and MCPR described in Section 4.3.10 during power operation.

4.4.6 Thermal Hydraulic Stability Description

Under certain conditions, BWRs can be susceptible to coupled neutronic/thermal-hydraulic instabilities. These instabilities are characterized by periodic power and flow oscillations and are the result of density waves (i.e., regions of highly voided coolant periodically sweeping through the core). If the power and flow oscillations become large enough, and the density waves contain a sufficiently high void fraction, the fuel cladding integrity safety limit could be challenged.

Types of Boiling Water Reactor Oscillations

There are two types of oscillations associated with BWR stability.

Type 1 instabilities experienced during startup do not result in a reactivity/power response. Type 1 oscillations are characterized by initiation of vapor production in the chimney region leading to a reduction in hydrostatic head in the chimney and a resultant core flow increase, which, in turn, could cause voids to collapse in the chimney. The BWRX-300 reactor goes through an unstable phase during startup. This type of oscillation is unavoidable in a natural circulation reactor because the unstable power/flow region must be crossed prior to establishing a steady two-phase voided region in the chimney; however, the magnitude of the flow oscillations is typically very small. As Type 1 oscillations do not result in a change in core moderator density, there is no power response and therefore no challenge to cladding integrity.

Type 2 oscillations are characterized by periodic power and flow oscillations and are the result of density waves (i.e., regions of highly voided coolant periodically sweeping through the core). In Type 2 instability, if the power and flow oscillations become large enough, and the density waves contain a sufficiently high void fraction, the fuel cladding integrity safety limit could be challenged.

Coupled neutronic-thermal-hydraulic instabilities, also known as density-wave instabilities, are safety concerns for BWRs. Three recognized modes of density-wave instability are core-wide (when the power and flow of all core channels oscillate in phase), regional (when the power and flow of half the core channels oscillate out-of-phase with the other half), and single channel flow instability (when the flow in a single channel oscillates accompanied by small power oscillations).

Design Criteria

The most limiting stability condition in the BWRX-300 normal operating region is at the rated power/flow condition. The BWRX-300 is designed so that the core remains stable throughout the entire operating domain. In the time domain analysis, decay ratio is defined as the ratio of the amplitude of the first two successive peaks. For the BWRX-300, the decay ratio is averaged over the first few peaks.

Conservative design criteria are imposed on the core-wide, regional, and single-channel decay ratios under all conditions of normal operation and anticipated transients. The limiting mode (i.e., highest decay ratio) for thermal-hydraulic instabilities (Type 2 oscillations) is the core-wide mode. The channel and regional modes are highly damped.

Stability Solution Design

The BWRX-300 design has features that result in stable behavior in normal operation and minimize the effects of potential oscillations in off-normal conditions include:

- Small Core:
  - The small core size and higher inlet orifice pressure drop of the BWRX-300 reduces the likelihood of regional mode instabilities. Conservative analyses are performed to confirm that regional mode oscillations are not possible and that core oscillations would be core- wide dominant. Any unacceptable core-wide oscillation is mitigated by the high flux scram.
  - Tighter neutronic coupling precludes regional mode oscillations
  - Core-wide oscillations are the dominant mode
- Natural circulation:
  - No recirculation pump trips that result in significant change from stable to unstable conditions
  - Loss of Feedwater Heating (LFWH) AOO impact on stability is mitigated by Selected Control Rod Rapid Insertion (SCRRI) operation for a feedwater temperature reduction of [[ ]]. The SCRRI function is described in PSR Ch. 7, Section 7.3. The loss of feedwater heating AOO analysis is described in PSR Ch. 15, Section 15.5.
- Tall chimney:
  - Increases volume of water
  - Increases driving head and natural circulation flow
  - Dampens oscillations

- Large downcomer area:
  - Reduces flow resistance
- High inlet orifice pressure drop:
  - Improves two-phase to single phrase pressure drop ratio
- Balanced feedwater temperature:
  - Neither thermal margins nor decay ratios are compromised
  - Minimizes normal operation inlet subcooling
- Less subcooling

[[ ]] The BWRX-300 has been established to provide margin to the decay ratio criterion.

4.4.7 Design Bases

The thermal and hydraulic design of the reactor core provides adequate heat transfer from the fuel to the various heat removal systems, i.e. Nuclear Boiler System (NBS), Shutdown Cooling System (SDC), and Isolation Condenser System (ICS) during normal operation and AOOs.

Loss of fuel rod cladding integrity is not expected during normal reactor operation and AOOs. To satisfy this requirement, the following design bases have been established for the thermal and hydraulic design of the reactor core.

Margin to the SAFDL is maintained during normal steady state operation when the MCPR is greater than the required OLMCPR and the MLHGR is maintained below the maximum LHGR limit(s). The steady state OLMCPR and Thermal Mechanical Operating Limit are established for the most limiting AOO and are analyzed in PSR Ch. 15, Section 15.5 including uncertainties that provide reasonable assurance that no fuel damage results during AOOs.

4.4.8 Critical Power Design Basis

The CPR is the ratio of the predicted critical power to the actual power of the particular fuel assembly, both evaluated at the same pressure, mass flux and inlet subcooling. The MCPR is defined as the minimum CPR for any fuel assembly within a core and is the figure of merit to represent the reactor thermal performance or margin.

The objective for normal operation and AOOs is maintaining nucleate boiling and precluding boiling transition.

The figure of merit confirming compliance with this objective is the CPR. The CPR is the ratio of the bundle power where at least one fuel rod point within the assembly experiences the onset of boiling transition to the operating bundle power. A calculated CPR of 1.0 corresponds to the best estimate value for the onset of boiling transition as determined by the product specific GEXL correlation (GEXL17 for GNF2).

CPR limits are specified for maintaining adequate margin to the onset of the boiling transition. Adequate margin is defined to be a 95 % probability at a 95 % confidence level that no fuel rods are susceptible to boiling transition. These limits are calculated based on the three-step process defined in the sections that follow.

Fuel Cladding Integrity Safety Limit

The Fuel Cladding Integrity Safety Limit is calculated so that no significant fuel damage occurs during normal operation and AOOs on a cycle-independent basis. The Fuel Cladding Integrity Safety Limit is defined as the MCPR that ensures there is a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition. This limit is also referred to as Safety Limit MCPR or $MCPR_{95/95}$. The value is dependent only on the fuel design and establishes a lower limit for the cycle-specific $MCPR_{99.9\%}$ value.

$MCPR_{99.9\%}$

The $MCPR_{99.9\%}$ is determined on a cycle-specific basis to support the determination of the Operating Limit MCPR (OLMCPR). The $MCPR_{99.9\%}$ ensures that 99.9% of the fuel rods in the core are not susceptible to boiling transition when considering the nuclear core design, plant system uncertainties, manufacturing uncertainties, and calculational uncertainties.

Operating Limit Minimum Critical Power Ratio

A cycle specific OLMCPR provides adequate assurance that the $MCPR_{99.9\%}$ is not exceeded during normal operation and AOOs. By operating with the MCPR at or above the OLMCPR, the Fuel Cladding Integrity Safety Limit for that plant is not exceeded during normal operation and AOOs. This operating limit is obtained by combining the maximum Delta Critical Power Ratio Over Initial Critical Power Ratio i.e. the change in CPR through the transient divided by the initial CPR value for the most limiting AOO and the $MCPR_{99.9\%}$. The significance of this parameter is that it measures the transient response and the maximum value from the AOOs is used in combination with the $MCPR_{99.9\%}$ to establish the OLMCPR on a cycle-specific basis.

4.4.9 Maximum Linear Heat Generation Rate Design Basis

The Maximum LHGR (MLHGR) bases are described in Section 4.3.5. Thermal mechanical operating limits ensure margin to design limits for circumferential cladding strain and centerline fuel temperature. The adequacy of LHGR limits is evaluated for the most severe AOOs providing assurance that no fuel damage results during these postulated events.

4.4.10 Void Fraction Distribution Design Basis

The void fraction in a BWR fuel bundle has a strong effect on the neutron flux and power (or fission rate) distribution. Accurate prediction of the void fraction is important for evaluating the performance of the reactor and fuel. The void fraction is evaluated using correlations based on the characteristic dimensions of the fuel bundle and hydraulic properties of the two-phase flow in the fuel bundle.

4.4.11 Core Pressure Drop and Hydraulic Loads Design Basis

An accurate model of core pressure drop is essential for modeling natural circulation flow, fuel and core inlet flow, and hydraulic loads for input to other evaluations. The total bundle pressure drop is defined as the sum of four components: friction, elevation, acceleration, and local losses. In analytical pressure drop models, the fuel assembly is divided into control volumes where the four components of total pressure drop are evaluated separately. This captures the effects on pressure drop of axially variable geometry parameters such as flow area, hydraulic diameter, wetted/heated perimeters, heat flux, and spacer elevations. The hydraulic diameter is defined as four times the axial flow area divided by the wetted perimeter, at any axial location and includes the fuel rod, channel inner wall, and water rod perimeters. The geometry of heated surfaces consists of the number of fuel rods and the fuel rod diameter in a fuel assembly. For fuel assembly types with partial length rods, the number of partial length rods and the associated length(s) are also accounted for in defining fuel assembly hydraulic diameter.

The TRACG methods for core pressure drop modeling are described in NEDE-32176P (Reference 4-14). The TRACG hydraulic formulation for core pressure drop is identical to the model used in the core design analysis except for the acceleration pressure drop component. The models used in the core design analysis are described in NEDC-34039P (Reference 4-13). The fuel design specific loss coefficients and assembly pressure drop models are developed from and confirmed by data from full scale testing of prototypical assemblies spanning the range of hydraulic conditions where hydraulic models are applied.

The adequacy of the pressure drop model applied to GNF2 fuel is summarized in GNF2 pressure drop characteristics and is inclusive of the BWRX-300 operating conditions in GNF2 pressure drop characteristics, NEDC-34040P, “BWRX-300 GNF2 Fuel Assembly Pressure Drop Characteristics,” (Reference 4-15).

Hydraulic loads are determined based on the reactor internal pressure differences. The TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs.

4.4.12 Core Coolant Flow Distribution Design Basis

Based on the prediction of core pressure drop, the distribution of flow into the fuel channels and the core bypass regions are calculated. The core coolant flow distribution forms the basis for predicting steady-state and transient MCPR and void fraction.

4.4.13 Fuel Heat Transfer Design Basis

Engineering models in both steady state and transient analysis tools predict heat transfer between fuel pellet, cladding gap, cladding, fuel rod surface and the coolant in the evaluation of core and fuel safety criteria.

4.4.14 Thermal-Hydraulic Stability Design Basis

The reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that SAFDL are not exceeded during any condition of normal operation, including the effects of AOOs.

The reactor core and associated coolant, control, and protection systems are designed to assure that power oscillations that could result in conditions exceeding SAFDL are either not possible or can be reliably and readily detected and suppressed.

4.4.15 Design Evaluation

4.4.16 Thermal and Hydraulic Evaluation Methods

Fuel Bundle Critical Power Method

The critical power is the fuel bundle thermal power at the onset of boiling transition. Maintaining the bundle power below the critical power during steady-state operation and AOOs precludes the onset of boiling transition and satisfies the SAFDL pertaining to heat transfer from the fuel to the coolant (i.e., the Fuel Cladding Integrity Safety Limit). The methods applied in determining the bundle critical power and the associated operating limits are described below.

The bundle critical power performance methodology was originally described in NEDO-10958-A SH 0001, “General Electric BWT Thermal Analysis Basis (GETAB) Data, Correlation and Design application, Licensing Report,” (Reference 4-16). This original methodology evolved into the current form of the correlation, i.e., the GEXL correlation. The GEXL correlation is a critical quality and boiling length correlation used to predict the occurrence of boiling transition in BWR fuel. Each fuel bundle design has a specific set of correlation coefficients developed from full-scale test data. The specific GEXL correlation applied in analyzing GNF2 for all BWR types, including BWRX-300, is designated GEXL17, NEDC-33292P, “GEXL17 Correlation for GNF2 Fuel,” (Reference 4-17). The GEXL17 correlation application range established for fleetwide application envelopes the hydraulic conditions that the BWRX-300 experiences during normal operation and AOOs.

The fuel cladding integrity safety limit, named the $MCPR_{95/95}$, ensures there is a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition using a limit that is derived from comparing the predicted critical power to the experimental data for a specific fuel bundle design. The Experimental CPR is defined as the ratio of the calculated critical power as determined by the GEXL correlation to the experimental critical power. Each experimental data point has a predicted value and associated Experimental CPR. The Experimental CPR is evaluated for all the points in the dataset resulting in a probability distribution. The Experimental CPR probability distribution serves as the basis for the correlation uncertainty. Thus, for a given critical power correlation, a limit that bounds 95% of a correlation's Experimental CPR distribution at a 95% confidence level is determined and set as the $MCPR_{95/95}$. The determination of the $MCPR_{95/95}$ is further described in TSTF-564-A, "Safety Limit MCPR," (Reference 4-18).

The cycle specific $MCPR_{99.9\%}$ limit is established as described in Appendix C of NEDC-34039P (Reference 4-13).

Maximum Linear Heat Generation Rate Method

The MLHGR methods are described in Section 4.2.5. Margin to design limits for circumferential cladding strain and centerline fuel temperature is evaluated for AOOs in accordance with NEDC-33840P (Reference 4-7).

Void Fraction Distribution Method

Empirical correlations are used for calculating the void fraction in the 3D core simulator and the steady-state thermal hydraulic calculations and underpin the correlations for the interfacial shear used in TRACG. The TRACG void fraction model is described in NEDE-32176P (Reference 4-14). The core simulator model is described in NEDC-34039P (Reference 4-13).

Core Pressure Drop and Hydraulic Loads Method

The total bundle pressure drop is defined as the sum of four components: friction, elevation, acceleration, and local losses. In these models, the bundle is also divided into control volumes where the four components of total pressure drop are evaluated separately. This allows capturing the effects on pressure drop of axially variable geometry parameters such as flow area, hydraulic diameter, wetted/heated perimeters, heat flux, and spacer elevations. The hydraulic diameter is defined as four times the axial flow area divided by the wetted perimeter, at any axial location and includes the fuel rod, channel inner wall, and water rod perimeters. The geometry of heated surfaces consists of the number of fuel rods and the fuel rod diameter in a fuel assembly. For fuel assembly types with partial length rods, the number of partial length rods and the associated length(s) are also accounted for in defining fuel assembly hydraulic diameter.

The TRACG methods for core pressure drop modeling are described in NEDE-32176P (Reference 4-14). The TRACG hydraulic formulation for core pressure drop is identical to the model used in the core design analysis except for the acceleration pressure drop component. The models used in the core design analysis are described in NEDC-34039P (Reference 4-13). The fuel design specific loss coefficients and assembly pressure drop models are developed from and confirmed by data from full scale testing of prototypical assemblies spanning the range of hydraulic conditions where hydraulic models are applied. The adequacy of the pressure drop model applied to GNF2 fuel is summarized in GNF2 pressure drop characteristics and is inclusive of the BWRX-300 operating conditions in NEDC-34040P (Reference 4-15).

Hydraulic loads are determined based on the reactor internal pressure differences. The TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs.

Core Coolant Flow Distribution Method

The core coolant flow distribution methods used in TRACG are described in Chapters 6 and 7 of NEDE-32176P (Reference 4-14). TRACG treats all fuel channels as one-dimensional (axial) components, but the vessel is modeled as a three-dimensional component. Hence, the pressure drop across two planes in the vessel is the same at all radial and azimuthal locations if the geometry of the components in the vicinity of these planes has radial and azimuthal symmetry. Otherwise, this pressure differential displays some (locally) radial and azimuthal non-uniformly. The flow distribution to the fuel assemblies and bypass flow paths in the core simulator model is calculated assuming the pressure drop across all fuel assemblies and bypass flow paths is the same. The bundle pressure drop evaluation includes frictional, local, elevation, and acceleration losses described above. The core inlet flow is an input to the core simulator. The value used in core design analysis is determined based on the TRACG prediction of the natural circulation core inlet flow. In operation, the core monitoring function determines core inlet flow based on plant instrumentation discussed in PSR Ch. 7, Section 7.3.

The bypass flow methodology is described in NEDE-32176P (Reference 4-14). The same methodology is used in the core simulator model.

Fuel Heat Transfer Method

The Jens-Lottes heat transfer correlation is used to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling as described in ANL-4627, "Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High-Pressure Water," (Reference 4-19). For the single-phase convection or liquid region, the Dittus-Boelter correlation is used. The methodology for fuel cladding, gap and pellet heat transfer is described in NEDC-33256P-A (Reference 4-4).

Thermal-Hydraulic Stability Analysis Method

TRACG is a GEH proprietary version of the Transient Reactor Analysis Code (TRAC). TRACG uses a multi-dimensional, two-fluid model for the reactor thermal-hydraulics and a three-dimensional reactor kinetics model. The models can be used to accurately simulate a large variety of test and reactor configurations. These features allow for realistic simulation of a wide range of BWR phenomena and are described in detail in NEDE-32176P (Reference 4-14).

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TRACG is qualified to accurately model natural circulation for a wide application range that encompasses the BWRX-300. The TRACG qualification bases detailed in NEDE-32177 (Reference 4-20) include benchmarking natural circulation flow and instability onset (FRIGG-4 FT-36C Onset of Instability Tests) in which TRACG conservatively predicts the power at the onset of limit cycle oscillations (over a range of system pressures and inlet subcooling) to be reasonably close to measured values.

Validity of Thermal and Hydraulic Design Techniques

The thermal and hydraulic design technique comprises qualified analytical methods employed in developing a self-consistent set of design outcomes that conform to design bases. The TRACG method described in Section 4.4.7 demonstrates accurate model system performance in BWRs. The thermal hydraulic design bases evaluated for the BWRX-300 described in Section 4.4.7 are applicable and adequate to those established for the operating fleet. The BWRX-300 reactor core was developed from applying qualified analytical methods and the results demonstrate compliance to the design bases that are used in BWRs.

4.4.17 Fuel Bundle Critical Power Evaluation

Compliance to representative steady-state MCPR operating limits is demonstrated for a typical simulation of the reference equilibrium cycle described in Section 4.3. The typical OLMCPR evaluation process is outlined in the sections that follow.

Fuel Cladding Integrity Safety Limit Evaluation

The GNF2 Fuel Cladding Integrity Safety Limit value is 1.07 as per TSTF-564-A (Reference 4-18).

$MCPR_{99.9\%}$ Evaluation

The $MCPR_{99.9\%}$ is evaluated for a specific core design with uncertainties documented in NEDC-34039P (Reference 4-13). The $MCPR_{99.9\%}$ limit is computed on a cycle-specific basis and reported in a cycle specific Core Operating Limits Report (COLR).

Operating Limit Minimum Critical Power Ratio Evaluation

The Operating Limit MCPR is computed on a cycle-specific basis and reported in a cycle specific COLR. The expected range of cycle-specific OLMCPRs for the reference BWRX-300 equilibrium cycle is depicted in Figure 4-9.

4.4.18 Maximum Linear Heat Generation Rate Evaluation

Compliance to steady-state MLHGR limits is demonstrated for the reference equilibrium cycle in Section 4.3 and NEDC-34160P (Reference 4-11). The AOO analysis for the reference equilibrium cycle is documented in PSR Ch. 15, Section 15.5. Compliance to design limits for circumferential cladding strain and centerline fuel temperature during AOO events are confirmed on a cycle-specific basis and associated limits are reported in the COLR.

4.4.19 Void Fraction Distribution Evaluation

The void fraction distribution is dependent upon the reactor state and varies throughout an operating cycle. The calculation of the void fraction distribution is integral to the TRACG methodology and the 3D core simulator, PANAC11 which is used to provide the axial variation of power (Figure 4-14). Representative values for the core average axial void fraction for are depicted in Figure 4-15.

4.4.20 Core Pressure Drop and Hydraulic Loads Evaluation

The expected operating pressure for the BWRX-300 is within the qualification basis of the pressure drop methods. The $MCPR_{99.9\%}$ calculation method also assumes pressure drop uncertainty.

4.4.21 Core Coolant Flow Distribution Evaluation

The core coolant flow distribution (i.e., the inlet flow to each fuel assembly) is determined by the coupled nuclear and thermal hydraulic steady-state methods in NEDC-34039P (Reference 4-13) and in the design and analysis of the reference core nuclear design summarized in Section 4.3.

4.4.22 Fuel Heat Transfer Evaluation

Fuel heat transfer evaluations are dependent upon the reactor state. The calculation of fuel heat transfer is integral to the TRACG methodology and the 3D core simulator, PANAC11.

4.4.23 Thermal-Hydraulic Stability Evaluation

To determine whether core-wide oscillation is the dominant mode, the flow velocity for symmetric channels on opposite sides of the core are perturbed out of phase. If the core is not susceptible to regional mode oscillations after such a flow velocity perturbation, then symmetric out of phase channels come into phase after a short duration. This analysis confirms that regional mode oscillations are not possible. As such, only the core-wide mode is applicable to the BWRX-300.

Stability Performance During Normal Operation

Stability analyses are performed at rated conditions and at multiple exposure points during the cycle. Stability analyses are performed by perturbing the system and assessing the core response. Such perturbations might include pressure pulse perturbation or variation in feedwater temperature. The response to a pressure perturbation in the steam line is analyzed to obtain the decay ratio for the BWRX-300 stability analysis.

The resulting decay ratios affirm stable operation, with margin to the stability decay ratio criteria, as described in PSR Ch. 15, Section 15.5.

Stability Performance During AOOs

In general, the stability margin reduces when the reactor power to flow ratio increases and/or core flow reduces or the core inlet subcooling increases (that also results in power increase). As the BWRX-300 is a natural circulation reactor, recirculation pump trip transients are not applicable and the key state variable that affects decay ratio is the inlet subcooling. As such, the loss of feedwater heating AOO is the limiting transient for stability.

The loss of feedwater heating event increases core inlet subcooling (i.e., the inlet temperature decreases) and increases core power. The BWRX-300 has been established to initiate a SCRRI in response to a loss of feedwater heating AOO that mitigates the increase in core thermal power. Stability analyses for the LFWH AOO with SCRRI affirm the stability decay ratio criterion is met and the results are summarized in PSR Ch. 15, Section 15.5.

4.5 Reactivity Control Systems Design

Reactivity Control within the BWRX-300 design consists of:

- Control rods and Control Rod Drive Systems (discussed within in this section)
- Supplementary reactivity control in the form of fuel rods containing gadolinia (discussed within Section 4.2)

4.5.1 Description

4.5.2 Control Rods Description

The control rods perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of control rods. These rods are positioned to counterbalance steam voids in the top of the core and effect significant power flattening. These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all control rods be available for either reactor "scram" (i.e., prompt shutdown) or reactivity control. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, electro-hydraulically actuated drive mechanisms that allow either electric motor controlled axial positioning for reactivity regulation or hydraulic scram insertion. The design of the rod-to-drive connection permits each control rod to be coupled or uncoupled from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and remain operable for tests with the reactor vessel open.

The core reactivity control requirements are met by use of the combined effects of the movable control rods, supplementary burnable neutron absorber (i.e., Gadolinia Rods in the fuel bundle described in Section 4.2), and the reactor coolant natural flow.

The control rod main structure consists of a top handle, an absorber section, and a bottom connector assembled into a cruciform shape. The top handle contains a grapple opening for handling. The absorber section is an array of stainless-steel tubes filled with boron carbide powder capsules or a combination of boron carbide powder capsules and hafnium rods. The connector is positioned on the bottom of the control rod for coupling to the control rod drive. While being inserted into the core, the control rod is restricted to the cruciform envelope created by the fuel bundles. Handle pads guide the control rod along the channels and connector rollers guide the control rod within the guide tube as the control rod is inserted and withdrawn from the core. The general configuration of the control rod is shown in Figure 4-16, whilst Figure 4-17 shows a cross-sectional view of the rod indicating the locations of the laser welds. Figure 4-18 provides details of the arrangement of an individual absorber tube.

The BWRX-300 employs the Ultra-HD control rod, which is based on the Ultra-HD control rod designed for the BWR/2 through BWR/6. This design has been applied to operating plants. The GEH Ultra-HD control rod is a derivative of prior designs incorporating hafnium rods in outer edge, high depletion tube locations. A detailed description is given in NEDE-33284-Supp 1-P-A-Rev 1, "Marathon-Ultra Control Rod Assembly," (Reference 4-22).

4.5.3 Control Rod Drive System Description

The CRD system includes three major elements:

- Electro-hydraulic FMCRD mechanisms
- HCU
- CRD hydraulic subsystem

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods.

The hydraulic power required for scram is provided by high pressure water stored in the individual HCUs. Each HCU contains a scram accumulator (nitrogen-water), charged to high pressure and the necessary valves and components to scram two FMCRDs. Additionally, during normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs.

The CRD hydraulic subsystem provides clean, demineralized water that is used to charge the scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD hydraulic subsystem is also the source of pressurized water for purging the SDC pumps and filling the NBS reactor water level reference leg instrument lines.

Fine Motion Control Rod Drive Mechanism

The Fine Motion Control Rod Drive (FMCRD) used for positioning the control rod in the reactor core is an electro-hydraulic actuated mechanism. An electric motor-driven ball nut and ball screw assembly positions the drive at both nominal increments and continuously over its entire range at a nominal speed. The FMCRDs also have the capability for motor-driven fast control rod insertion. The FMCRD penetrates the bottom head of the RPV. The FMCRD does not interfere with refueling and is operative even when the head is removed from the RPV.

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The FMCRD design provides an anti-rotation device that engages when the lower component is removed for maintenance. This device prevents rotation of the ball screw and hence prevents control rod motion when the lower component is removed. The anti-rotation device consists of:

- The coupling piece on the bottom of the ball screw that engages with the lower component drive shaft
- The back seat of the middle flange

The coupling between the lower component drive shaft and ball screw is splined to permit removal of the lower housing. The underside of the coupling piece on the ball screw has a circumferentially splined surface that engages with a mating surface on the middle flange backseat when the ball screw is lowered during lower component removal. When engaged, ball screw rotation is prevented. In the unlikely event of total failure of all the drive flange bolts attaching the lower component flange and middle flange of the drive to the housing flange, the anti-rotation device can engage when the lower component falls. The middle flange/outer tube/CRD blowout support is restrained by the control rod guide tube base bayonet coupling, thus preventing rod ejection.

Magnetic Coupling

The magnetic coupling is located at the bottom of the lower component. It is employed to achieve leak-free operation of the FMCRD without seals. The magnetic coupling consists of an inner and an outer rotor. The inner rotor is located inside the lower component pressure boundary. The outer rotor is located outside the pressure boundary. Each rotor has permanent magnets mounted on it. As a result, the inner and outer rotors are locked together by magnetic forces acting through the pressure boundary and work as a synchronous coupling. The outer rotor is coupled with the motor unit and driven by the motor such that the inner rotor follows the rotation of the outer rotor.

The magnetic coupling is designed so that its maximum coupling torque exceeds the maximum torque of the motor unit to prevent decoupling or slippage due to motor torque.

The magnetic coupling is designed to not have any undesirable effects on other magnetic sensitive sub-components.

Materials of Construction

The materials of construction for the FMCRD components are selected for compatibility with the reactor coolant, wear resistance, corrosion resistance and material strength to ensure reliability and design life requirements are met in the BWRX-300 environment.

Hydraulic Control Units

Upon receipt of a scram signal, each HCU furnishes pressurized water for hydraulic scram to two FMCRD units (except for the FMCRD in the center of the core which has its own HCU). Additionally, each HCU provides the capability to adjust purge flow to the two drives. A test port is provided on the HCU for connection to a portable test station to allow for controlled venting of the scram insert line to test the FMCRD ball check valve during plant shutdown. The check valves shown inside the HCU boundary function to close under system pressure, fluid flow and temperature conditions during scram. The check valves ensure that the water stored in the HCU accumulator is delivered to the FMCRDs to accomplish the scram function.

A simplified single line diagram for the HCU is provided in Figure 4-20.

Scram Solenoid Valve Assembly

The scram solenoid valve assembly consists of one valve and two solenoids which control the position of the scram valve. The solenoid valves are normally energized and closed. Upon loss of power to the solenoids, the valves open which vents air to open the scram valve. The assembly is designed so that the power must be removed from both solenoids before air pressure can be discharged from the scram valve operator. This prevents the inadvertent scram of the drives associated with a given HCU in the event of a failure of one of the valve solenoids.

Scram Valve

The scram valve opens to supply pressurized water to the bottom of the drive piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the MCR as soon as the valve starts to open.

Scram Accumulator

The scram accumulator stores sufficient energy to fully insert two control rods at any anticipated reactor pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging line prevents loss of water pressure in case charging water header pressure is lost. During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. A pressure transmitter provides local and MCR nitrogen pressure indication. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and raises an alarm in the DCIS. The alarm would prompt operator action to repressurize the nitrogen bottle using an external supply of nitrogen gas. To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A level sensor actuates an alarm in the MCR if water leaks past the piston barrier and collects in the accumulator instrumentation block.

Purging Panel

The purging panel controls the purge water flow to the associated FMCRDs. Each panel has a needle valve in the purge water line to control the purge water flow rate. This orifice maintains the flow at a constant value while the drives are stationary. A bypass line containing a solenoid-operated valve is provided around this orifice. The valve is signaled to open and increase the purge water flow whenever either of the two associated FMCRDs is commanded to insert by the RC&IS. During FMCRD insertion cycles, the hollow piston moves upward, leaving an increased volume for water within the drive. Opening of the purge water makeup valve increases the purge flow to offset this volumetric increase and precludes the backflow of reactor water into the drive, thereby preventing long-term drive contamination.

Control Rod Drive Hydraulic Subsystem

The CRD hydraulic subsystem consists of two sets of equipment and supplies clean demineralized water to the following:

- HCU accumulators for charging
- FMCRDs for purge water
- SDC pumps for seal purge water
- NBS reactor water level reference leg instrument lines
- PRM for sampling

The subsystem consists of two trains, each providing the required functions with the necessary pumps, valves, filters, piping, and instrumentation. A simplified single line diagram for the CRD sub-system is provided in Figure 4-21.

4.5.4 Design Bases

4.5.5 Control Rods Design Basis

The control rods are designed to control the fission chain reaction. The rods, along with the control rod drive system provide stable and automatic control of reactor core power, spatial instabilities, and local power density during normal operation. The control rods also shut down the reactor and maintain the core subcritical.

The control rod design meets the following acceptance criteria:

- Control rod stresses, strains, and cumulative fatigue are evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection
- The control rod design is evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses
- Control rod materials are shown to be compatible with the reactor environment
- Control rod reactivity worth is included in the plant core analyses

The following bases are established for the above acceptance criteria.

Stress, Strain, and Fatigue

The control rod design is evaluated to assure that it does not fail because of loads due to shipping, handling, normal operation, including the effects of AOOs, Postulated Accident (PA) and Design Extension Condition (DEC). To ensure that the control rods do not fail, these loads must not exceed the ultimate stress and strain limit of the material, structure, or welded connection. Fatigue must not exceed a fatigue usage factor of 1.0. It is found that the calculated fatigue usage is less than the material fatigue capability (the fatigue usage factor is much less than 1.0).

The loads evaluated include those from normal operational transients (scram and rod maneuvering), pressure differentials, thermal gradients, flow, and system induced vibration, and irradiation growth in addition to the lateral and vertical loads expected for each condition. Fatigue usage is based upon the cumulative effect of the cyclic loadings. The analyses include corrosion and crud deposition as a function of time, as appropriate.

Conservatism is included in the analyses by including margin to the limit or by assuming loads greater than expected for each condition. Higher loads can be incorporated into the analyses by increasing the load itself or by statistically considering the uncertainties in the value of the load.

Control Rod Insertion

The control rod design is evaluated to assure that it can be inserted during normal operations including the effects of AOOs, PAs and DECs. These evaluations include a combination of analyses of the geometrical clearance and actual testing. The analyses consider the effects of manufacturing tolerances, swelling and irradiation growth.

Control Rod Material

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Reactivity

The reactivity worth of the control rod design is determined by the initial amount and type of absorber material and irradiation depletion. Scram time insertion performance must also be included in the plant core analyses including the effects of normal operations, AOOs, PAs and DECs.

4.5.6 Control Rod Drive System Design Basis

The CRD system provides the primary means of reactivity control during normal, abnormal and accident conditions. The system design basis includes two diverse motive forces for the CRD insertion (scram) using high pressure water from the HCUs, and control rod insertion using the FMCRD motor. Incorporated into the design are positioning and protective features that prevent inadvertent withdrawal, drop, and ejection of the control rod due to a component break or other malfunction.

4.5.7 Design Evaluation

4.5.8 Control Rods Evaluation

The control rod design is evaluated against the acceptance criteria and bases described in Section 4.5.5. Design compliance with these criteria constitutes the basis for acceptance and approval of the design. The control rods for the BWRX-300 are based on the design used in the operational BWR fleet. This well understood fleet operational data is utilized in the methods to design, evaluate, and analyze the control rods.

Scram

The largest axial structural loads on a control rod blade are experienced during a control rod scram, due to the high terminal velocity. To be conservative, structural analyses of the control rod are performed assuming a 100% failed CRD buffer. A dynamic model is used to simulate a detailed representation of the load bearing components of the assembly during a scram event. Simulations are run at atmospheric temperatures, pressures, speeds, and properties as well at operating temperatures, pressures, speeds, and properties.

Seismic

Fuel channel deflections which result from seismic events impose lateral loads on the control rods NEDE-33284 (Reference 4-22). The BWR/2 through BWR/6, ABWR and BWRX-300 have similar channel lengths and deflections. As a result, control rods in these BWRs experience similar lateral deflection load and resultant component stresses and strains.

The seismic analysis is performed by evaluating the strain in the Ultra-HD control rod absorber section when deflected. During a seismic event, it is assumed the seismic deflections could be added to any preexisting channel bow. The absorber section strain has been analyzed for channel deflections due to seismic and channel bow deflections when deflected by a bounding value and found to be acceptable as described in NEDE-33284 (Reference 4-22).

Testing was performed on the ABWR Marathon control rod to confirm seismic scram capability. The ABWR Marathon control rod was tested at fuel channel oscillation amplitudes representing design bases earthquake conditions. The scram times were found to be acceptable, and the Ultra High Duty control rod was not damaged as described in NEDE-33284 (Reference 4-22). Because the Ultra High Duty geometry and bending stiffness are nearly identical to the tested Marathon, the ABWR Marathon control rod seismic scram capability testing is applicable to BWRX-300 conditions.

Stuck Rod

In the event that a control rod becomes stuck in a fuel cell due to friction with the fuel channels, a scram would impose large axial compression loads on the control rod. The control rod is evaluated to withstand this large axial load without buckling or component failure, even if the entire load is applied to one wing as described in NEDE-33284 (Reference 4-22).

Absorber Irradiation Related Loads

The absorber containment licensed in NEDE-33284 (Reference 4-22) is applicable to the BWRX- 300 Ultra HD control rod. The same methodology is used for the BWRX-300 Ultra HD control rod in NEDE-33284 (Reference 4-22). The absorber tube and B_4C capsule design accommodates irradiation-induced swelling of boron carbide, such that a clearance exists between the inner capsule and outer absorber tube up to the maximum local B-10 depletion. The analysis conservatively assumes worst-case dimensions plus a bounding boron carbide irradiation swelling rate. As such, all stresses on the outer absorber tube associated with boron carbide swelling are eliminated, which drastically reduces the likelihood of Irradiation-Assisted Stress Corrosion Cracking relative to previous BWR control rod designs. Absorber tube stresses due to helium gas generation and moisture vapor heating are considered. The resulting stresses due to helium pressure are small compared to tube material strengths. Therefore, the Ultra HD absorber containment design is adequate for the nuclear design life of the control rod.

Load Combinations and Fatigue

The BWRX-300 Ultra-HD control rod is designed to withstand load combinations including AOOs and fatigue loads associated with those combinations. Absorber tube loads are evaluated during a scram, in a cell with severe channel bow near end of control rod life when absorber burn-up helium gas generation is highest. Absorber section to connector welds and absorber section to handle loads are evaluated during a scram when the absorber helium gas build up is highest. In accordance NEDE-33284 (Reference 4-22), the BWRX-300 Ultra-HD control rod does not exceed the ultimate stress or strain limit of the material. Based on the reactor cycles, the combined loads are then evaluated for the cumulative effect of the cyclic loadings NEDE-33284 (Reference 4-22). The fatigue usage is evaluated against a limit of 1.0.

Handling Loads

The BWRX-300 Ultra-HD control rod is designed to accommodate three times the weight of the control rod, NEDE-33284 (Reference 4-22).

Hydraulic Loads

The Ultra High Duty control rod is not damaged by the vibrations or cavitations set up by coolant velocities and velocity distributions in the bypass region between fuel channels.

Materials

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Nuclear Performance

The nuclear lifetime of the initial BWRX-300 control rod type is established as a 10% reduction in reactivity worth (Δk-effective/k-effective) relative to the initial reactivity worth in any quarter axial segment, NEDE-33284 (Reference 4-22).

Mechanical Compatibility

Similar to the control rods supplied for the ABWR and BWR/2 through BWR/6, the BWRX-300 Ultra-HD control rod is designed to be compatible with core and reactor internal interfaces.

The control rod coupling socket provides a compatible interface with the FMCRD. The coupling engages the FMCRD by rotating. With the FMCRD, Control Rod Drive Housing (CRDH), and Control Rod Guide Tube positively assembled, any orientation of the cruciform control rod between the fuel assemblies is a coupled position, and rotation to an uncoupled position is not possible during reactor operation. The four lobes of the FMCRD coupling spud are in line with the four wings of the control rod in the coupled position.

The control rod is designed to permit coupling and uncoupling of the control rod drive from below the vessel for FMCRD servicing without necessitating the removal of the reactor vessel head. The control rod is also designed to allow uncoupling and coupling from above the vessel using control rod handling tools.

The control rod is dimensionally compatible with the fuel assemblies (unirradiated and irradiated).

4.5.9 Control Rod Drive System Evaluation

Scram Time

The control rod scram function of the CRD system provides the negative reactivity insertion required by the Safety Design Bases in Section 4.5.6. The required scram time provided in PSR Ch. 15, Table 15.5-5 is used in the safety analyses.

Scram Reliability

Key features resulting in high scram reliability for the CRD system include:

- Scram valves open by spring action and are normally held closed by pressurized control air.
- To cause hydraulic scram, a de-energizing reactor trip signal is provided to the solenoid-operated pilot valve that vents the control air from the scram valves for opening.
- The SC1 I&C hydraulic scram is designed so that the HCU scram signal independently initiates a hydraulic scram demand, from whatever source, regardless of any other rod positioning signal.
- The FMCRD hollow piston and guide tube are designed so they do not restrain or prevent control rod insertion during scram.
- Each FMCRD mechanism initiates electric motor-driven insertion of its control rod simultaneous with the initiation of hydraulic scram upon receipt of I&C signal. This provides a diverse means to assure control rod insertion.
- The system is “fail-safe” in that loss of either electrical power to the scram solenoids, or loss of control air pressure to the scram valve operator, causes a scram.
- Departure from the ball-nut releases spring-loaded latches in the hollow piston to engage slots in the guide tube:
    - These latches support the hollow piston in the fully inserted position.
    - Following a hydraulic scram insertion, the control rod cannot be withdrawn until the ball-nut is driven up, re-engaged, and the hollow piston de-latched from the guide tube.

- The design also includes ARI pilot valves on the control air header, which serves all 29 scram valves:
  - The ARI pilot valves are energized-to-actuate and provide an alternate path to vent control air and open all scram valves resulting in hydraulic insertion of all control rods.

4.6 Core Monitoring System Design

Core monitoring is a function of the plant computer system that provides three-dimensional core power monitoring. Core monitoring provides confidence that the plant is operating in conformance with SAFDL. Core monitoring obtains instrumentation information from the Distributed Control and Information System, calculates power distributions and resulting thermal limits. These power distributions are adapted to signals from plant GTs and LPRMs as applicable.

4.6.1 Description

The core monitoring function acquires live reactor data from site plant data acquisition systems as required, to define the reactor state for use by the core simulator. The acquired data are validated within acceptable ranges. On a periodic basis, upon user request, or when triggered by a system event, the core monitoring system determines current core characteristics based on the current reactor state and the previous history of the following core components:

- Thermal limit monitoring
- Soft duty guideline monitoring
- Gamma Thermometer (GT) processing
- Prediction at achievable operating regimes
- Tracking of local and global xenon behavior
- Identification of LPRM drift
- Power adaption based on in-core power measurement
- Graphical display of data
- Power / flow tracking
- BWR operating guidelines implementation
- Isotopic tracking

Core parameters are calculated either by the core simulator or during post processing (including parameters associated with fuel bundles, fuel channels, GTs, LPRMs, and control rods). The core monitoring function then compares these core parameters against technical specifications, licensing limitations, manufacturer's guidelines, and site-specific limits. If user actions are required, the system sends alerts, warnings, and notifications to the Plant Alarm System. The primary core monitoring functions are:

- Calculation of current core parameters
- Prediction of future core parameters
- Calculation of LPRM calibration factors
- Calculation and tracking of the isotopic inventory of the fuel
- Generation of visualizations and reports on core performance
- Generates core technical information for use by other plant systems as required

The core monitoring function uses live reactor and in-core information, and coupled with a three-dimensional BWR simulator model, produces current and predicted core performance information. When the live plant input data, including LPRM, control rod position, and core power are not available, the core monitoring function still performs a core performance calculation using the Manual Monitor option. The performance information includes standard visualizations such as thermal-hydraulic parameters, thermal margins, fuel conditioning margins, fuel burnup, LPRM information, control rod position information and power/flow maps. Information from the core monitoring function is used by the Licenced Plant Operator and the Reactor Engineer in the reactor operational decision-making process, for both steady-state operation and during plant maneuvers, to ensure compliance with licenced limits.

Core monitoring also supports the calibration process for LPRMs by processing GT data and live LPRM data to yield individual LPRM calibration factors. The core monitoring function does not directly calibrate the LPRMs, instead the calibration adjustments require operator action to be implemented in the LPRM systems.

The core monitoring function uses a coupled nuclear thermal-hydraulic diffusion theory model, to provide a three-dimensional simulation of BWR core characteristics performance. The accuracy of the modeling is enhanced by adaptive algorithms that conform results to measured plant data from the LPRMs and GTs. Instrument signals are compared to simulator predictions and outlier data are rejected as anomalous. Unavailability or failure of a limited number of LPRM and GT signals does not significantly affect plant operation.

4.7 References

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4-2 NEDE-33798P, “Application of NSF to GNF Fuel Channel Designs,” Revision 1, GE-Hitachi Nuclear Energy, Americas, LLC.

4-3 NEDE-24011-P-A, “General Electric Standard Application for Reactor Fuel (GESTAR II),” Revision 32, GE-Hitachi Nuclear Energy, Americas, LLC.

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4-5 NEDC-33257P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 2 – Qualification,” Revision 2, GE-Hitachi Nuclear Energy, Americas, LLC.

4-6 NEDC-33257P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 3 – Application Methodology,” Revision 2, GE-Hitachi Nuclear Energy, Americas, LLC.

4-7 NEDC-33840P, “The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance,” Revision 1, GE-Hitachi Nuclear Energy Americas, LLC.

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4-10 NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” Revision 11, GE-Hitachi Nuclear Energy, Americas, LLC.

4-11 NEDC-34160P, “Reactor Core Nuclear Design Report,” Revision 0, GE-Hitachi Nuclear Energy, Americas, LLC.

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4-15 NEDC-34040P, “BWRX-300 GNF2 Fuel Assembly Pressure Drop Characteristics, Revision 0, GE-Hitachi Nuclear Energy, Americas, LLC.

4-16 NEDO-10958-A SH 001, “General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design application, Licensing Topical Report,” GE-Hitachi Nuclear Energy, Americas, LLC., January 1977.

4-17 NEDC-33292P, “GEXL17 Correlation for GNF2 Fuel,” Revision 3, GE-Hitachi Nuclear Energy, Americas, LLC.

4-18 Technical Specifications Task Force, “Safety Limit MCPR”, TSTF-564-A, Revision 2.

4-19 Argonne National Laboratory, “Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High-Pressure Water”, ANL-4627, 1951.

4-20 NEDE-32177, "TRACG Qualification", Revision 3, GE-Hitachi Nuclear Energy, Americas, LLC.

4-21 NEDC-33083, "TRACG Application for ESBWR," Revision 2, GE-Hitachi Nuclear Energy, Americas, LLC.

4-22 NEDE-33284-Supp 1- P-A-Rev 1, "Marathon-Ultra Control Rod Assembly," Revision 2, GE-Hitachi Nuclear Energy, Americas, LLC.

4-23 Office for Nuclear Regulation, "Safety Assessment Principles for Nuclear Facilities," 2014 edition, Revision 1.

4-24 NEDC-34140P, "BWRX-300 Safety Case Development Strategy," Revision 0, GE-Hitachi Nuclear Energy, Americas, LLC.

4-25 NEDC-34159P, "Fuel Summary Report," Revision 0, GE-Hitachi Nuclear Energy, Americas, LLC.

Table 4-1: Analytical Techniques used in Core Design

Analysis	Technique	Computer Code
Fuel Rod Design	Numerical solutions of 1D steady-state and transient heat transfer and finite element mechanical analysis	PRIME03P
Fuel Performance Characteristics (temperature, internal pressure, clad stress, etc.)		
Nuclear Design Cross-sections and group constants	Lattice physics	TGBLA06
X-Y and X-Y-Z power distribution, reactivity coefficients	Steady-state coupled nuclear thermal-hydraulics Quasi 2-group diffusion theory	PANAC11
Axial power distributions, control rod worths		
Fuel rod power		
Thermal-hydraulic design steady-state	Multi-dimensional, two-fluid model thermal-hydraulics 3D reactor kinetics	TRACG

Notes:

1D: One-Dimensional

3D: Three-Dimensional

TRACG: Transient Reactor Analysis Code General Electric

Table 4-2: GNF2 Fuel Assembly Key Attributes and Materials

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Table 4-3: Typical Thermal–Hydraulic Design Characteristics of the Reactor Core

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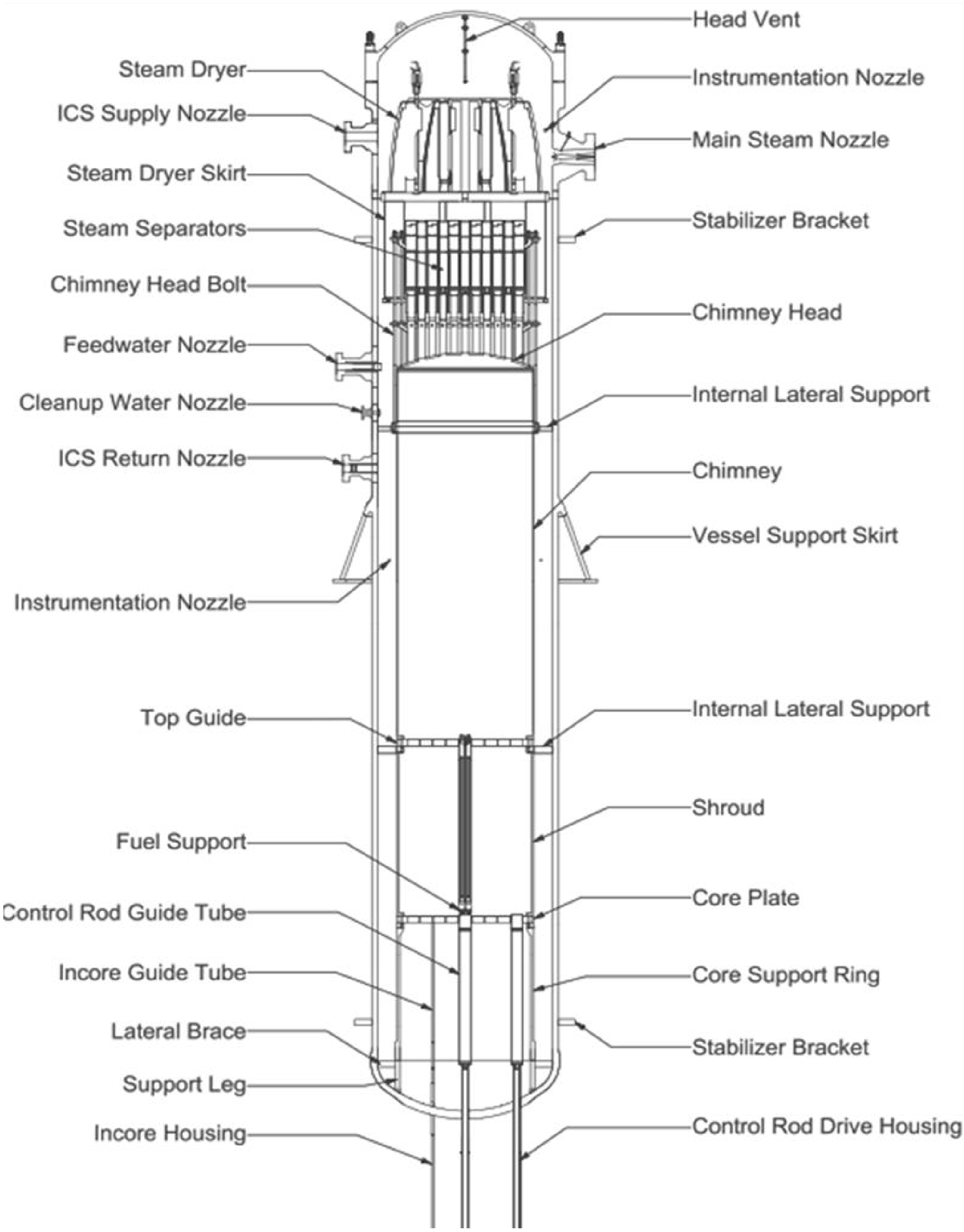

Figure 4-1: BWRX-300 Reactor Pressure Vessel and Internals

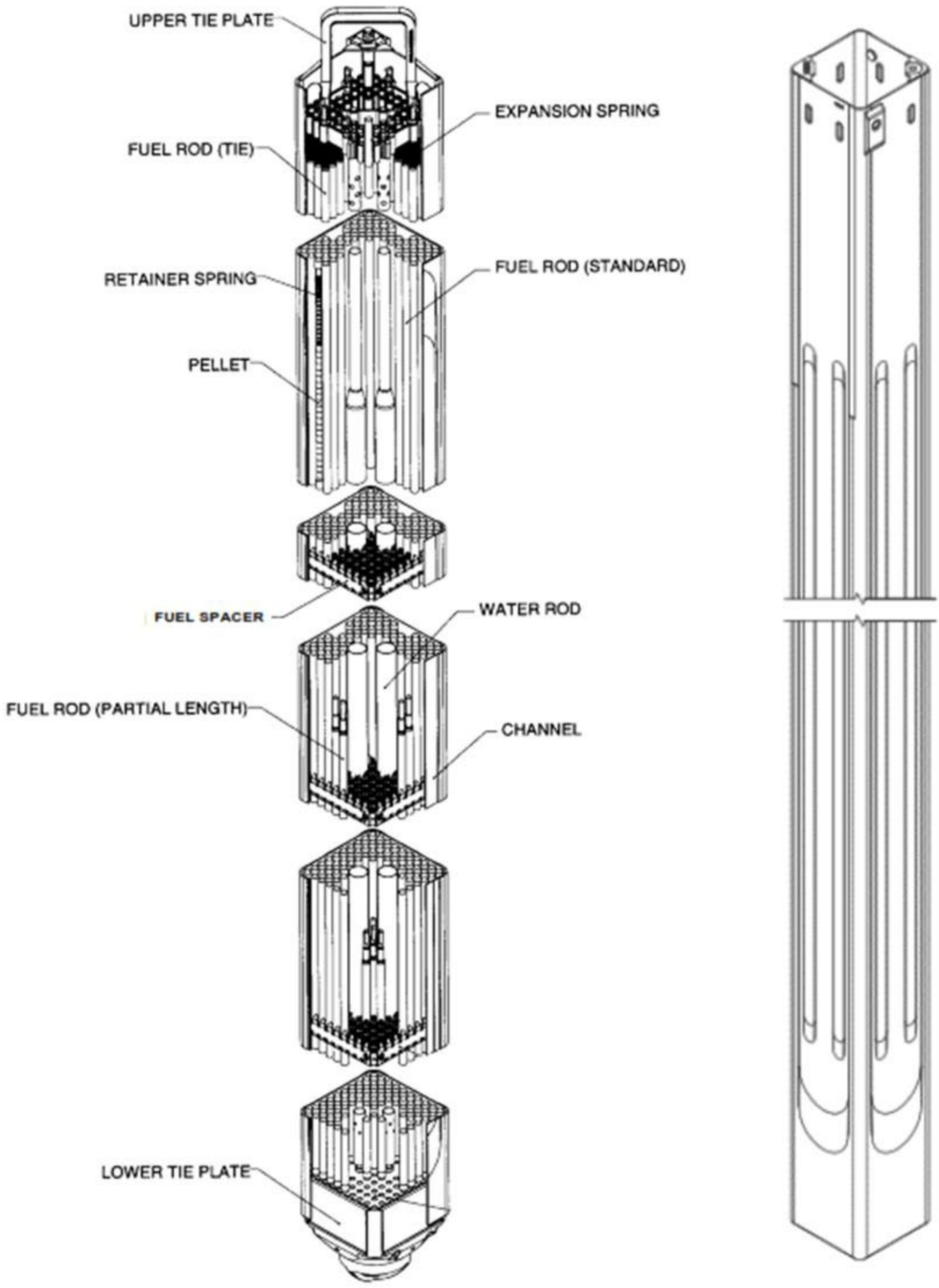

Figure 4-2: GNF2 Fuel Bundle and Channel

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Figure 4-3: GNF2 Lattice Array

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Figure 4-4: Core Arrangement (Including Preliminary Instrumentation Layout)

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Figure 4-5: Reference Equilibrium Cycle Core Loading/Shuffling Pattern

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Figure 4-6: Control Rods used for Reactivity Control during the Reference Equilibrium Cycle

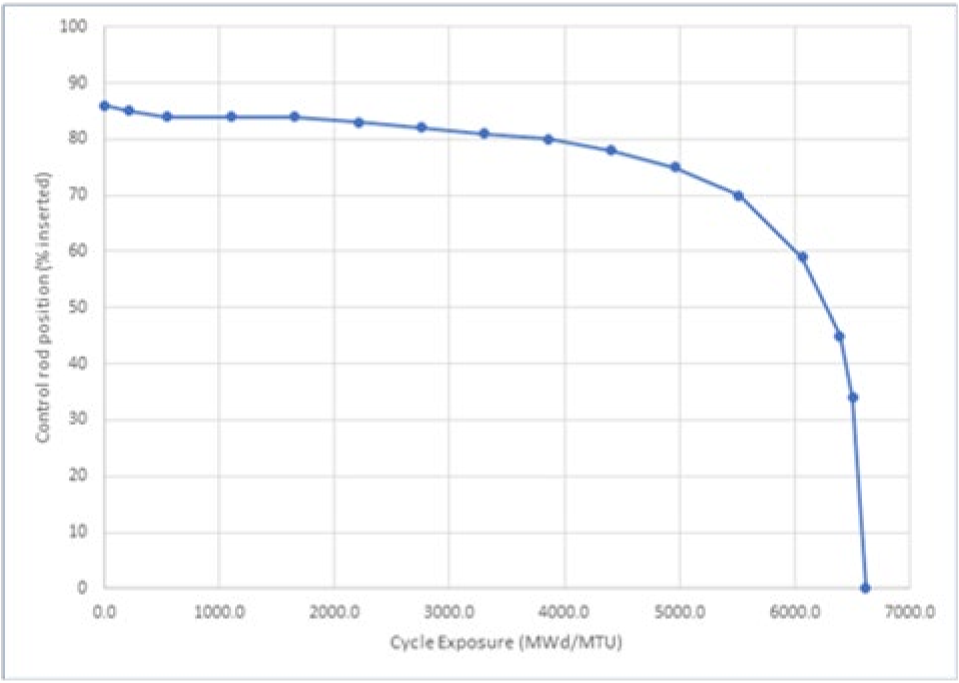

Figure 4-7: Control Rod Insertion vs Cycle Exposure for the Reference Equilibrium Cycle

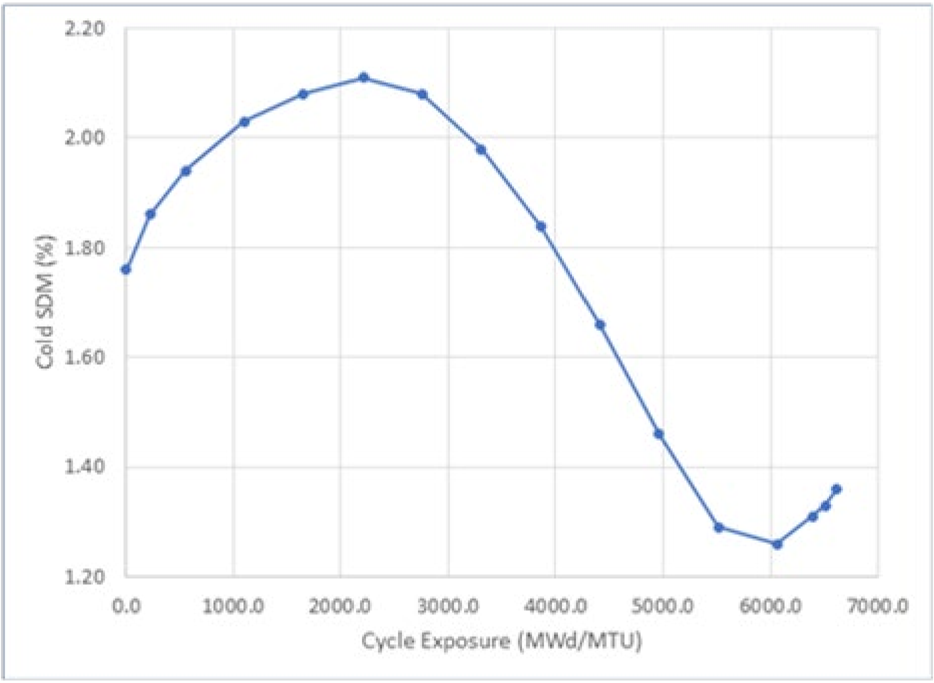

Figure 4-8: Cold Shutdown Margin vs. Cycle Exposure for the Reference Equilibrium Cycle

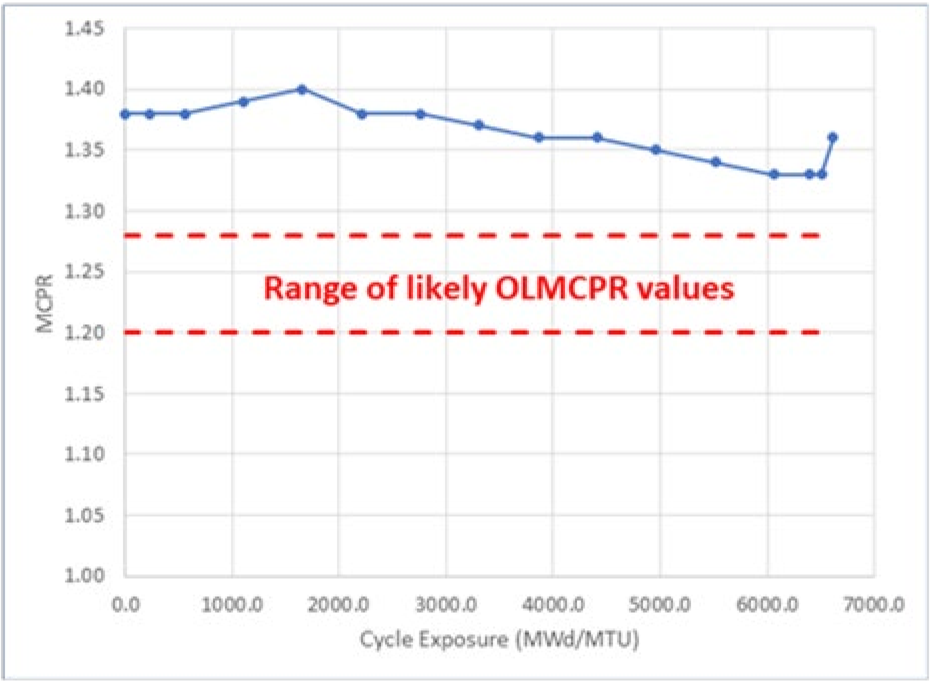

Figure 4-9: MCPR vs. Cycle Exposure for the Reference Equilibrium Cycle

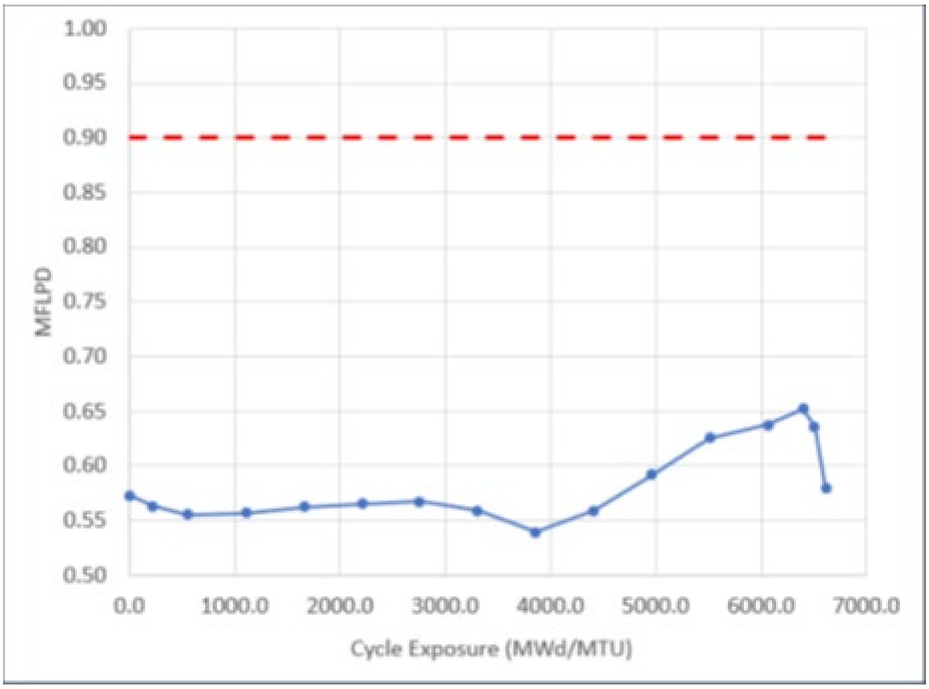

Figure 4-10: MFLPD vs. Cycle Exposure for the Reference Equilibrium Cycle

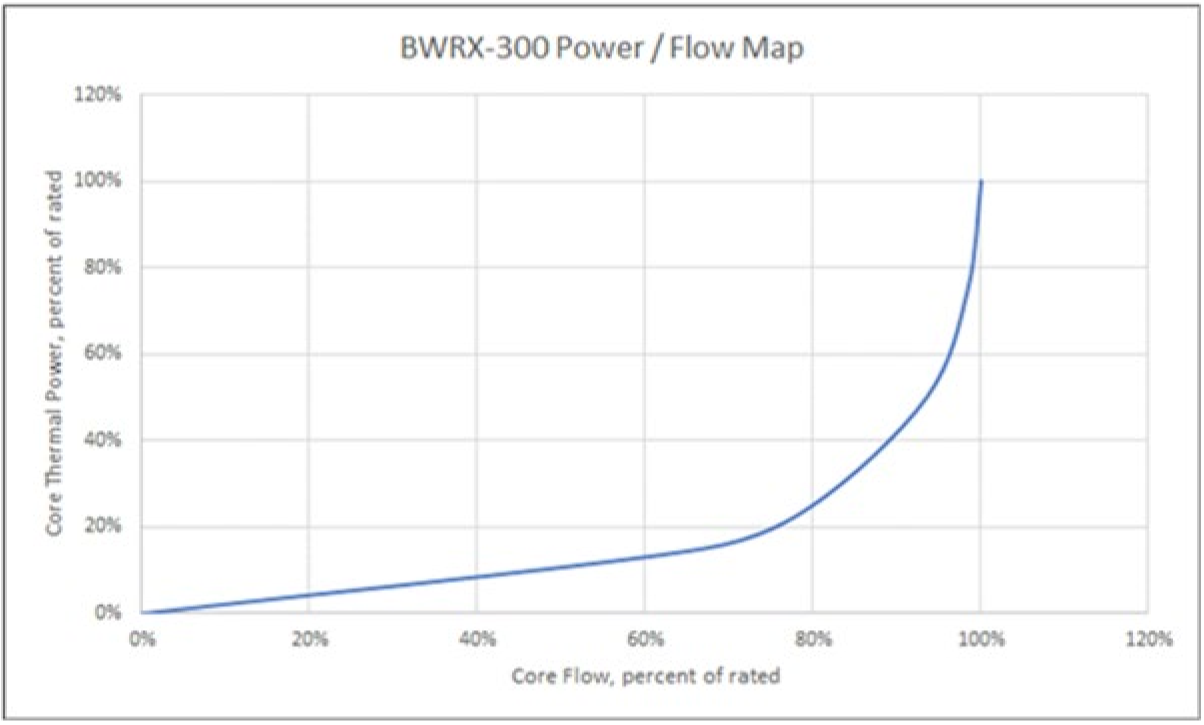

Figure 4-11: BWRX-300 Power/Flow Map

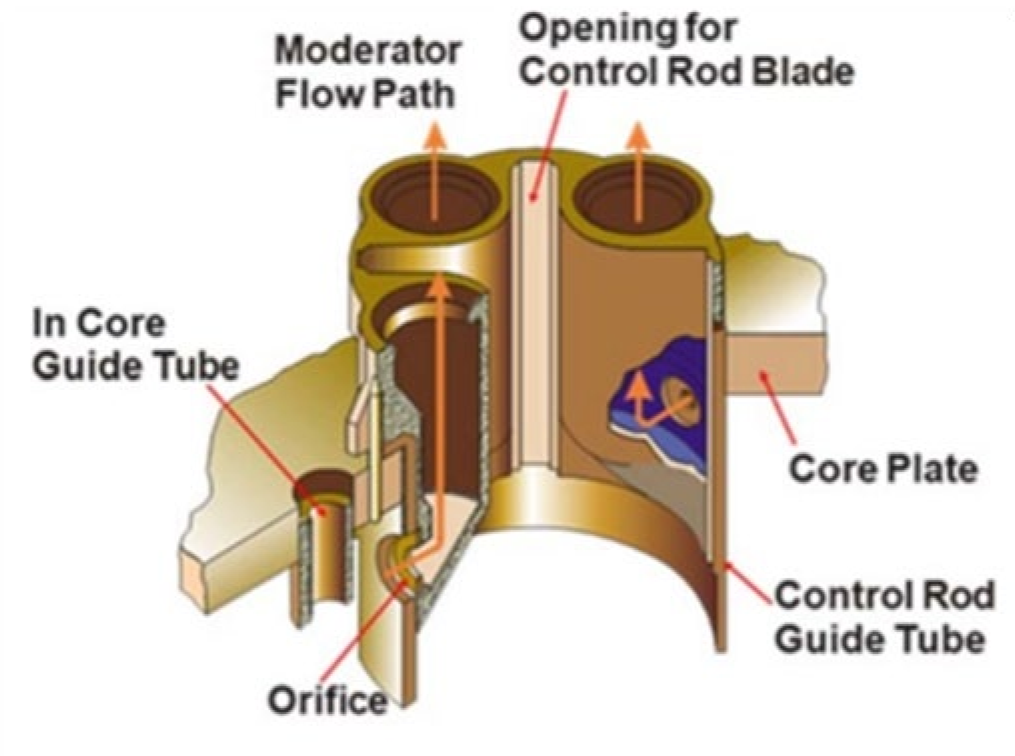

Figure 4-12: Orificed Fuel Support

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Figure 4-13: BWRX-300 Core Inlet Orifice Type Arrangement

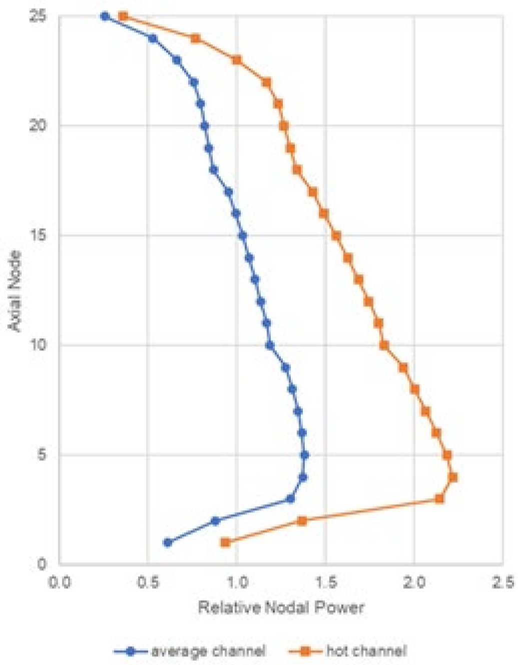

Figure 4-14: Relative Power for Analyzed Node (hot channel and average channel)

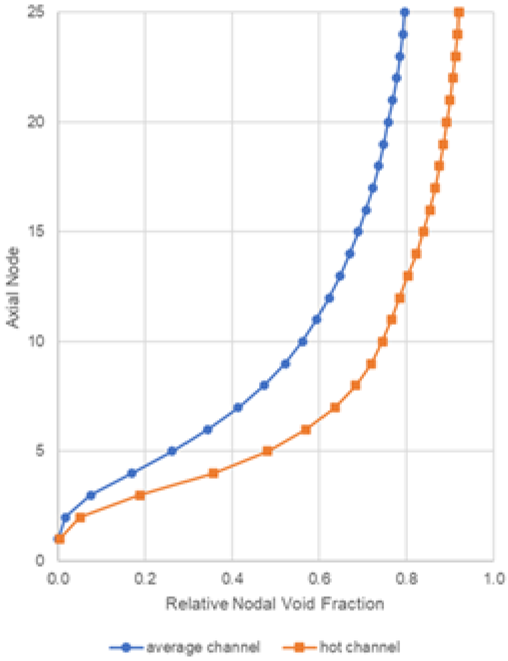

Figure 4-15: Relative Void Fraction for Analyzed Node (hot channel and average channel)

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Figure 4-16: Ultra-HD-Ultra Control Rod

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Figure 4-17: Cross-sectional View of Ultra-HD-Ultra Control Rod

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Figure 4-18: Ultra-HD-Ultra Control Rod Absorber Tube

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Figure 4-19: Schematic Diagram of FMCRD

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Figure 4-20: HCU Simplified Piping & Instrumentation Diagram (P&ID)

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Figure 4-21: CRD Subsystem Simplified Piping and Instrumentation Diagram

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

A.1 Claims, Arguments and Evidence

The Office for Nuclear Regulation (ONR) Safety Assessment Principles (SAPs) 2014 ("Safety Assessment Principles for Nuclear Facilities," (Reference 4-23) identify ONR's expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The CAE approach can be explained as follows:

- Claims (assertions) are statements that indicate why a facility is safe
- Arguments (reasoning) explain the approaches to satisfying the claims
- Evidence (facts) supports and forms the basis (justification) of the arguments

The GDA CAE structure is defined within the Safety Case Development Strategy (SCDS) (NEDC-34140P, "BWRX-300 Safety Case Development Strategy," (Reference 4-24) and is a logical breakdown of an overall claim that:

> *"The BWRX-300 is capable of being constructed, operated and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK".*

This overall claim is broken down into Level 1 claims relating to environment, safety, security, and safeguards, which are then broken down again into Level 2 area related sub-claims and then finally into Level 3 (chapter level) sub-claims.

The Level 3 sub-claims that this chapter demonstrates compliance against are identified within the SCDS, NEDC-34140P (Reference 4-24) and are as follows:

2.1.1 The safety functions (Design Basis) and integrity claims have been derived for the fuel and associated reactor core systems through a robust analysis, based upon Relevant Good Practice (RGP).

2.1.2 The design of the fuel and associated reactor core systems has been substantiated to achieve the required safety functions in all relevant operating modes.

2.1.3 The design of the fuel and associated reactor core systems has been undertaken in accordance with proven methodologies, analysis tools and design safety principles and taking account of Operational Experience (OPEX) to support reducing risks to (ALARP).

2.1.4 The performance of the fuel and associated reactor core systems are validated by suitable testing, inspection and monitoring throughout manufacturing, operation, and site-based storage.

2.1.5 Ageing and degradation mechanisms applicable to the fuel and associated reactor core systems are identified and assessed as an integral part of the design process. Suitable examination, inspection and monitoring are specified to ensure the integrity of fuel remains fit-for-purpose through-life.

In order to facilitate compliance, demonstration against the above Level 3 sub-claims, this PSR chapter has derived a suite of arguments that comprehensively explain how their applicable Level 3 sub-claims are met (see Table A-1).

It is not the intention to generate a comprehensive suite of evidence to support the derived arguments, as this is beyond the scope of GDA Step 2. However, where evidence sources are available, examples are provided.

A.2 Risk Reduction As Low As Reasonably Practicable

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a 2-Step GDA. It is considered that the most that can be realistically achieved is to provide a reasoned justification that the BWRX-300 Small Modular Reactor (SMR) design aspects could effectively contribute to the development of a future ALARP statement. In this respect, this chapter contributes to the overall future ALARP case by demonstrating that:

- The chapter-specific arguments derived may be supported by existing and future planned evidence sources covering the following topics:
  - RGP has demonstrably been followed. A significant information source for the development of the RGP applied to the design of the BWRX-300 reactor core is the operational experience information obtained from the worldwide inservice performance of GNF2 fuel as provided in Appendix A to NEDC-34159P, “Fuel Summary Report,” (Reference 4-25).
  - OPEX has been taken into account within the design process
  - All reasonably practicable options to reduce risk have been incorporated within the design
- It supports its applicable level 3 sub-claims, defined within the SCDS, NEDC-34140P (Reference 4-24)

Probabilistic safety aspects of the ALARP argument are addressed within PSR Ch. 15.

Table A-1: PSR Chapter 4 (Reactor) Claims and Arguments

Level 3 Chapter Claim	Chapter 4 Argument	Sections and/or Reports that Evidence the Arguments
2.1 The design and operation of the fuel and core has been derived and substantiated taking into account RGP and OPEX.		
2.1.1 The safety functions (Design Basis) and integrity claims have been derived for the fuel and associated reactor core systems through a robust analysis, based upon RGP.	The PSR chapter specifies the design bases for the fuel, nuclear core design, thermal-hydraulic design, control rods (including control rod drive mechanism) and core monitoring system	Design bases are defined in the following sections of PSR Ch. 4. • 4.2.2 (fuel) • 4.3.2 (core nuclear design) • 4.4.2 (thermal hydraulic design) • 4.5.2.1 (control rods) • 4.5.2.2 (CRDM) • 4.6.2 (core monitoring system) RGP for fuel derived from OPEX obtained for GNF2 (and predecessor fuel designs) which is contained within Appendix A of NEDC-34159P (Reference 4-25) (Fuel Summary Report) Core nuclear design and thermal-hydraulic designs are informed by the in-service performance of actual fuel cycles operated by BWRs (and loaded with GNF fuel) operating throughout the world. RGP for control rod designs is expressed in NEDE-33284 (Reference 4-22)

<table>
<tr><th>Level 3 Chapter Claim</th><th>Chapter 4 Argument</th><th>Sections and/or Reports that Evidence the Arguments</th></tr>
<tr><td colspan="3">2.1 The design and operation of the fuel and core has been derived and substantiated taking into account RGP and OPEX.</td></tr>
<tr><td>2.1.2 The design of the fuel and associated reactor core systems has been substantiated to achieve the required safety functions in all relevant operating modes.</td><td>Substantiation of the fuel and associated core systems (control rods and CRDM) has been achieved through rig testing during fuel development and confirmed through demonstration of satisfactory in-service performance</td><td>Design evaluation/substantiation is provided in the following sections of PSR Ch. 4;
• 4.2.3 (fuel)
• 4.3.3 (core nuclear design)
• 4.4.3 (thermal hydraulic design)
• 4.5.3.1 (control rods)
• 4.5.3.2 (CRDM)
Ongoing substantiation of the design of the fuel is through the satisfactory in-service performance of GNF2 fuel via OPEX obtained of GNF2 (and predecessor fuel designs) in BWR reactors throughout the world. The OPEX is contained within Appendix A of NEDC-34159P (Reference 4-25) (Fuel Summary Report).</td></tr>
<tr><td>2.1.3 The design of the fuel and associated reactor core systems has been undertaken in accordance with proven methodologies, analysis tools and design safety principles and taking account of OPEX to support reducing risks ALARP</td><td>Design evaluation with emphasis on methodologies, codes and their validation.
GNF makes use of previously licenced (in other regimes) and validated methodologies and tools (computer programs) to undertake design evaluations of fuel designs and core configurations</td><td>Discussion of analytical methodologies applied to fuel, core and thermal hydraulic design are discussed in the following sections of PSR Ch. 4:
• 4.2.2 (fuel)
• 4.3.2 (core nuclear design)
• 4.4.2 (thermal hydraulic design)
The references listed below provide the basis, qualification, and application of methodologies for:
• Fuel thermal-mechanical design as embodied in the PRIME computer code (References 4-4 to Reference 4-7).
• Reactor core neutronic and thermal-hydraulic design as embodied in the TGBLA, PANAC and TRACG computer codes (Reference 4-14).</td></tr>
</table>

Level 3 Chapter Claim	Chapter 4 Argument	Sections and/or Reports that Evidence the Arguments
2.1 The design and operation of the fuel and core has been derived and substantiated taking into account RGP and OPEX.		
2.1.4 The performance of the fuel and associated reactor core systems are validated by suitable testing, inspection and monitoring throughout manufacturing, operation, and site-based storage.	Extensive test and inspection of fuel and associated reactor components is undertaken at point of manufacture as part of Quality Control activities. Inspection of fuel and associated reactor components is undertaken upon receipt at sites prior to loading into the reactor. Various programmes of in-service inspection are undertaken on site	Discussion of inspection and testing activities applied to fuel and associated reactor core systems is provided in the following sections of PSR Ch. 4: • 4.2.4 (fuel) • 4.5.4 (control rods) Validation of the in-service performance of GNF2 fuel is derived from OPEX obtained for operation of GNF2 (and predecessor fuel designs) in BWR reactors throughout the world. The OPEX is contained within Appendix A of NEDC-34159P (Reference 4-25) (Fuel Summary Report).
2.1.5 Ageing and degradation mechanisms applicable to the fuel and associated reactor core systems are identified and assessed as an integral part of the design process. Suitable examination, inspection and monitoring are specified to ensure the integrity of fuel remains fit-for-purpose through-life.	Inspection and monitoring programmes are established by BWR operators / users of GNF fuel. The results from these programmes are used to determine root causes and mechanisms for any in-service fuel failures/degradation detected.	RGP associated with ageing and in-service degradation mechanisms applicable to fuel is derived from OPEX obtained for operation of GNF2 (and predecessor fuel designs) in BWR reactors throughout the world. The OPEX is contained within Appendix A of NEDC-34159P (Reference 4-25) (Fuel Summary Report). In particular, this includes an analysis of fuel failure types/mechanisms. Development of appropriate site-based in-service inspection and monitoring programme for fuel is a Forward Action Plan item

APPENDIX B FORWARD ACTION PLAN

This appendix contains the Forward Action Plan for the reactor core topic area (see Table B-1). The plan provided contains the future work commitments and recommendations for future work where gaps to GDA expectations have been identified. A delivery phase for the completion of the identified work is also provided. These delivery phases are broadly aligned to; within the first two steps of the GDA, during development of the Pre-Construction Safety Report (PCSR) and prior to Site Licence application.

Table B-1: PSR Chapter 4 (Reactor) Forward Actions

Number	Finding	Forward Actions	Lead Discipline	Delivery Phase
PSR4-15	The design of the fuel bundles and core loading pattern for the initial cycle is currently not available. GNF are scheduled to have this information available for the various BWRX-300 projects in 2025.	Update Reactor Core Nuclear Design Report with details pertaining to the proposed initial core design and reference this in the pre-construction safety report.	Fuel & Core	During PCSR development
PSR4-16	Details of the features and capabilities of the actual core monitoring system (such as ACUMEN) is considered to be beyond the requirements of a two-step GDA. However, the requirements and basic functionality of a core monitoring system are included in PSR Ch. 4	Within the pre-construction safety report make reference to suitable documentation that describes the functionality, features, and capabilities of GEH's ACUMEN system.	Fuel & Core	During PCSR development
PSR4-17	Design Bases are considered within PSR Ch. 4, however detailed tabulation of all the safety functional claims pertaining to the fuel and core (including associated systems such as control rods, instrumentation) is a subject for the next phase of Licensing	Include a tabulation within the pre-construction safety report of the safety functional claims pertaining to the fuel and core including associated systems, if appropriate	Fuel & Core	During PCSR development

Number	Finding	Forward Actions	Lead Discipline	Delivery Phase
PSR4-18	Definition of LCOs is considered to be beyond the requirements of a two-step GDA.	Include a section within the pre-construction safety report which details the of the Limiting Conditions of operation applicable to the fuel and core, including associated systems	Fuel & Core	During PCSR development
PSR4-19	The management of failed fuel is considered to be beyond the requirements of a two-step GDA and is an operational rather than a design issue. However, it should be noted that there are no aspects of the design of the BWRX-300 that preclude the adoption of standard BWR fuel management strategies	Include a section within the pre-construction safety report which details the management of failed fuel for the BWRX-300. This should include detection, identification and in-core and ex-core control of fuel bundles with one or more leaking fuel rods. A dedicated reference providing full details of failed fuel management should be produced in a similar manner as was produced for Step 4 of the United Kingdom (UK) ABWR GDA	Fuel & Core Spent Fuel Management	During PCSR development
PSR4-20	Load following capability is considered to be beyond the requirements of a two-step GDA and is rather a future operator's choice whether they select that capability. If the operator selects load following capability, the BWRX-300 would be demonstrated to be able to comply with all applicable targets as part of site-specific licensing	Add information pertaining to load following capability of the BWRX-300 within the pre-construction safety report, if the future operator decides to take advantage of this capability.	Fuel & Core	Before Site License Application