# 4 REACTOR

The staff reviewed Chapter 4 of the CRN-1 PSAR, against applicable regulatory requirements using regulatory guidance and standards to assess the sufficiency of the preliminary information on the reactor design for the issuance of a construction permit (CP) in accordance with Title 10 *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities.” As part of this review, the staff evaluated information on the reactor design, with special attention given to design and operating characteristics, unusual or novel design features, and principal safety considerations. The staff evaluated the preliminary design of the reactor design to ensure the design criteria, design bases, and information relative to construction is sufficient to provide reasonable assurance that the final design will conform to the design basis.

The staff’s reviews and evaluations for areas relevant to PSAR Chapter 4, including regulations and guidance used, a summary of the application information reviewed, and evaluation findings and conclusions, are discussed in the safety evaluation (SE) sections below for each specific review area.

## 4.1 Summary Description

The staff reviewed Chapter 4 of the PSAR, which describes the BWRX-300 reactor design, including the RPV and internal components, fuel assembly and control rod design, nuclear design bases, and thermal-hydraulic methodologies. The chapter also addresses CRD system design features and thermal-hydraulic stability.

### 4.1.1 Reactor Pressure Vessel

Section 4.1 of the PSAR is a summary description of the purpose of the reactor pressure vessel (RPV) and highlights several unique features of the BWRX-300 design including the use of integral Reactor Isolation Valves (RIVs) and an increased vessel height to support natural circulation. The staff’s assessment of the RPV design, including use of integral RIVs, is described in Section 5 of this report; thermal hydraulic considerations related to the natural circulation design are discussed in Sections 4.5, 4A, and 15 of this report.

### 4.1.2 Reactor Internal Components

This section of the PSAR provides a summary description of the reactor internal components (also known as reactor internals). With the exception of the chimney structure, these are generally standard components found in existing, licensed BWRs (e.g., core support structures, control rods, steam separator and steam dryer). The applicant states that other than the Zircaloy in the reactor core, the reactor internals are stress corrosion-resistant stainless steels or other high alloy steels. Additional details regarding reactor internal materials are can be found in Section 4.5.2 of this SER.

The applicant specifies in PSAR Section 4.1.2.2 that the core support structures are designed, fabricated, and examined in accordance with the provisions of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPVC), Section III, Division 1, “Rule for Construction of Nuclear Facility Components,” Subsection NG, “Core Support Structures.” In PSAR Section 4.1.2.6.5, the applicant specifies surveillance sample holders are designed, fabricated, and analyzed to the requirements of ASME BPVC, Section III, Subsection NG.

### 4.1.3 Reactor Core

PSAR Section 4.1.3 describes the fuel assembly and control rod arrangement in the core. The core is comprised of 240 fuel assemblies arranged in control cells, which are sets of four fuel bundles separated into quadrants by cruciform control rod element. In addition, 12 fuel assemblies are located on the outer edge of the core and are not immediately adjacent to a control rod but are supported by peripheral fuel supports.

### 4.1.4 Reactivity Control Systems

This subsection of the PSAR provides cross references to the PSAR sections in which the reactivity control systems are described in detail. The fuel and control rod design is laid out in PSAR Section 4.2 and assessed in Section 4.2 of this report. The control rod drive (CRD) system is described in PSAR Section 4.6 and assessed in Section 4.6 of this report.

### 4.1.5 Nuclear Instrumentation

PSAR Section 4.1.5 describes the nuclear instrumentation system, including the use of fixed neutron detectors. Specifically, the CRN-1 core is instrumented with local power range monitors and gamma thermometers. Gamma thermometers at fixed locations are used in lieu of traversing in-core probes in BWR/2 through BWR/6 designs. A core monitoring system relying on gamma thermometers was approved for use in the ESBWR in NEDC-33197P-A, “Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring,” Revision 3. The applicant confirmed during the audit that NEDC-33197P-A provides the basis for core monitoring approach relying on gamma thermometers at the CRN-1 plant, and that CRN-1 gamma thermometers are designed and will be operated consistent with the description provided in NEDC-33197P-A, Revision 3, except for modifications that will be submitted to NRC staff for review with the operating license application (OLA) as documented during the audit (ML26091A346). This issue is further described in Section 4.3 of this report. Further NRC staff evaluation of core instrumentation is provided in Chapters 7 and 15 of this report.

PSAR Section 4.1.5 states that wide-range neutron monitors (WRNMs) are provided. It describes location and sensitivity of WRNMs. PSAR Section 7.3.3.2 states that WRNMs are used by both the operator and the automatic power regulator when starting the reactor. PSAR Section 7.3.3.2 also notes that WRNMs can provide a scram signal on reactor period when the reactor mode switch is in “startup.” NRC staff noted that accident and transient analysis provided in PSAR Chapter 15 only considered events initiated from rated full-power conditions.

### 4.1.6 Analysis Techniques

PSAR Section 4.1.6.1, “Neutron Fluence,” references NEDC-32983P-A, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations,” Revision 2. The NRC staff approved this methodology to provide best-estimate predictions of reactor pressure vessel neutron fluence for BWR/2 through BWR/6 designs (i.e., BWR designs licensed in the U.S. at the time of review and approval). Review of the final calculation of CRN-1 RPV fluence using the methodology in NEDC-32983P-A will be considered during review of the OLA.

## 4.2 Fuel System Design

Section 4.2 of the CRN PSAR provides an overview of the GNF2 fuel system design for the BWRX-300 design, detailing analysis methods, and the establishment of specified acceptable fuel design limits (SAFDLs).

PSAR Section 4.2 summarizes the analysis requirements that cover fuel system damage, fuel rod damage, and core coolability. Fuel system damage mechanisms encompass all components within the fuel assembly and are applicable to normal operation, including the effects of Type 1 anticipated operational occurrences (AOOs) which credit DL2 mitigation and use the SAFDLs as the acceptance criteria (Type 1 and Type 2 AOOs[1] are defined in Section 15.1.3.1 of this report). Fuel rod failure mechanisms are specific to the fuel rod and cladding and are associated with normal operation, AOOs, and postulated accidents. Finally, fuel coolability applies to the fuel assembly retaining its rod-bundle geometry during postulated accidents.

Operating limits are established to ensure that actual fuel operation is maintained within the fuel rod thermal-mechanical design and safety analysis bases. These operating limits define the maximum allowable fuel pellet operating power level as a function of fuel pellet exposure. The applicant's analyses cover each damage or failure mechanism and provide the applicable SAFDLs and a concluding summary of the ability of the PSAR fuel system design, based on the GNF2 fuel assembly, to meet these limits.

The applicant states that the GESTAR II licensing framework (NEDE-24011-P-A-31-US) will be used to define the generic requirements and approved methods for BWR fuel design. This is documented in NEDC-3370P, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)" (ML070780344). The general approach applied by the applicant is to define an operational envelop in terms of peak nodal linear heat generation rate (LHGR) vs. peak nodal burnup where safe operation at steady state has been demonstrated. Regarding AOOs, the applicant further defines thermal over-power (TOP) and mechanical over-power (MOP) screening limits to evaluate whether Type 1 AOOs will cause a cladding strain limit or pellet temperature limit to be exceeded. If a Type 1 AOO exceeds the thermal or mechanical over-power, the PRIME methodology (see Section 4.2.3) will be used to analyze that AOO relative to the applicable SAFDL.

The sections below document the information provided by the applicant, and the review conducted by the staff, for each of the PSAR topics.

### 4.2.1 Fuel Assembly Design Bases

#### *4.2.1.1 Introduction*

The design and safety objectives of the fuel system are to ensure that fuel design limits will not be exceeded during normal operations or Type 1 AOOs and that the effects of Type 2 AOOs and postulated accidents will not cause significant damage to the fuel and reactor coolant pressure boundary (RCPB) or impair the capability to cool the core.

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[1] As discussed in Section 15.1.3.1 of this report, Type 1 analysis credits DL2 mitigation of the AOO initiating event, while the Type 2 analysis considers only DL3 mitigation of the AOO in conjunction with failure of DL2 functions. SAFDLs are used as the acceptance criteria for Type 1 AOO analysis, while Type 2 analysis uses the acceptance criteria in the "safety-related 10 CFR 50.2 definition.

*4.2.1.2 Regulatory Evaluation*

The following NRC regulations are the relevant requirements for the BWRX-300 SMR fuel design:

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," describes requirements for analyzing cooling performance of the emergency core cooling system (ECCS), using an acceptable evaluation model, and establishing acceptance criteria for light-water nuclear power reactor ECCSs.

10 CFR 50, Appendix A, General Design Criterion (GDC) 10, "Reactor design," as it relates to ensuring that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

GDC 26, "Reactivity Control System Redundancy and Capability," as related to the requirement for at least two independent reactivity control systems of different design, and requirements on the capability to ensure SAFDLs are not exceeded under normal operation and AOOs with appropriate margin for malfunctions such as stuck rods.

GDC 27, "Combined reactivity control systems capability," as it relates to the reactivity control systems being designed with appropriate margin and, in conjunction with the ECCS, being capable of controlling reactivity to maintain the capability of cooling the core under postulated accident conditions.

GDC 35, "Emergency core cooling," as it relates to designing the reactor fuel system such that the performance of the ECCS will not be compromised following a postulated accident.

The guidance in Section 4.2, "Fuel System Design," of NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition* (SRP), lists the acceptance criteria that are adequate to meet the above requirements and review interfaces with other SRP sections. As noted in PSAR Table 1.9-4, "Conformance with NUREG-0800 (Chapter 4 Reactor)," the applicant states that the PSAR conforms to SRP 4.2 acceptance criteria applicable to the CP application.

*4.2.1.3 Technical Evaluation*

SAFDLs are established by the applicant to show compliance with the stated NRC regulations and design bases. The technical evaluation of BWRX-300's fuel system design bases, SAFDLs, and the methods used to establish them is provided later in this Section 4.2.3 of this report.

*4.2.1.4 Conclusion*

The conclusion regarding BWRX-300's fuel system SAFDLs and the methods to establish them is provided in Section 4.2.3 of this report.

### 4.2.2 Fuel Assembly Description

*4.2.2.1 Introduction*

PSAR Section 4.2.2, "Fuel Assembly Description," describes the fuel system design, as summarized, in part, below for the BWRX-300 reactor.

The PSAR states that the GNF2 fuel assembly design will be used in the BWRX-300 reactor and that there are no design differences in the GNF2 fuel assembly for the BWRX-300 reactor relative to the GNF2 fuel that is currently used in existing BWRs. GNF2 is a 10×10 fuel assembly design with full length and partial length fuel rods, water rods, spacer grids, and the fuel channel. Upper and lower tie plates are connected by fueled tie rods that support the weight of the bundle during fuel handling. The NSF alloy (a GNF proprietary zirconium-based alloy) channel has extensive operating experience in operating BWRs (e.g., see NEDE-33798P, "NSF Channel Annual Experience Summary Report" Supplement 1, Revision 7 [ADAMS Package Accession No. ML24012A008]). The operating experience has shown good performance with respect to fluence gradient-induced bow and shadow corrosion-induced bow. The PSAR lists the various functions of the GNF2 channel, which include thermal-hydraulic, structural, and fuel sipping (fuel sipping is a method to identify if rod[s] in a fuel assembly have lost hermeticity) functions.

As stated in PSAR Table 4.2-1 and NEDC-3370P, the GNF2 fuel rods consist of uranium dioxide ($UO_2$) cylindrical ceramic pellets and a round wire compression spring located in the plenum, encapsulated within a cladding of recrystallized Zircaloy-2 tube with an inner barrier liner of natural zirconium. The fuel rods are internally pressurized with helium during assembly.

The fuel rod plenum, which is located above the pellet column, allows space for axial thermal differential expansion of the fuel column and accommodates the initial helium loading and evolved fission gases. The plenum spring at the top of the fuel pellet column keeps the column in its proper position during handling and shipping.

The applicant stated that gadolinia burnable poison is added to some of the fuel rods to reduce the excessive reactivity in fuel that has higher enrichment in order to control power peaking. In the poisoned fuel rods, gadolinium oxide ($Gd_2O_3$) is mixed with the $UO_2$.

#### *4.2.2.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.2.1.2 of this report also applies to this subsection.

#### *4.2.2.3 Technical Evaluation*

The NRC staff finds that the fuel system was described in appropriate detail in the PSAR. The fuel design is an existing licensed fuel design. The evaluation of the fuel system design is described in Section 4.2.3 of this report.

#### *4.2.2.4 Conclusion*

The NRC staff concluded that the fuel system design was appropriately described; the adequacy of the fuel design is evaluated in Section 4.2.3 of this report.

### **4.2.3 Fuel Assembly Design Evaluation**

#### *4.2.3.1 Introduction*

PSAR Section 4.2.3 states that the GESTAR II licensing framework (NEDE-24011-P-A-31-US) will be used to define the generic requirements and approved methods for BWR fuel design. During the audit the staff requested the applicant provide a list of deviations from GESTAR II for BWRX-300. In response to this audit request, the applicant stated that the GESTAR II methodology would be used without any modifications. The only difference in these evaluations is in plant and reactor-specific inputs such as inputs to the fatigue analysis.. Based on a review of NEDC-34042P, “BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report” (ML26097A239), the staff concluded that none of the preliminary inputs for the BWRX-300 design are outside the ranges of input approved for GESTAR II. Section 4.2.3 of the PSAR specifies that the main fuel rod thermal-mechanical design analysis is performed using the PRIME code and methodology as described in, NEDC-33256P-A, “The PRIME Model for Analysis of Fuel Rod Thermal - Mechanical Performance Part 1 - Technical Bases;” NEDC-33257P-A, “The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance Part 2 - Qualification;” and NEDC-33258P-A, “The PRIME Model for Analysis of Fuel Rod Thermal - Mechanical Performance Part 3 – Application Methodology for steady-state operation.” The applicant notes that the PRIME methodology is approved for application to AOO transients in NEDC-33840P-A, “The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance.” The prior NRC reviews of PRIME and other GNF analysis methods did not consider the effect of load follow operation. However, PSAR Section 4.2.3.8 includes consideration of daily load follow operation in the fuel cladding fatigue analysis. Because the staff considers there could be additional aspects of daily load follow operation that could impact the PRIME methods, the staff will review operation of BWRX-300 under conditions of daily load following and justification of the applicability of fuel thermal-mechanical analysis methods for analyzing the conditions during review of the OLA if the CRN-1 plans to employ load following operation.

Additionally, in order to ensure that the licensing basis is clearly established and can be maintained during operation of the facility, at the OLA phase, the staff will verify that FSAR includes a listing of the nuclear codes, methods and analytical results (within either technical or topical reports) that are either incorporated by reference, or are described, outlined or summarized in sufficient detail to be considered described within FSAR for the purpose of applying 10 CFR 50.59, as described in Section 3.10 of NEI 96-07, Revision 1.

GVH previously submitted generic fuel rod thermal-mechanical analyses for the GNF2 fuel assembly in NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)” (ML070780344), in 2007. The NRC previously audited NEDC-33270P in its evaluation of Amendment 32 and 33 to the U.S. Supplement to GESTAR II. NEDC-34042P, “BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report” (ML26097A239) summarizes the preliminary PRIME thermal-mechanical design analyses for GNF2 fuel in the BWRX-300. The staff compared the fuel thermal-mechanical preliminary steady state and Type 1 AOO acceptance criteria provided in NEDC-34042P and observed that the preliminary acceptance criteria were consistent with those provided in NEDC-33270P and/or GESTAR II. The staff noted that some of the primary differences in NEDC-34042P and NEDC-33270P include the fatigue analysis and different fuel residence time for BWRX-300, which would affect phenomena such as cladding corrosion.

Similarly, preliminary GNF2 fuel assembly mechanical evaluations for the BWRX-300 were provided in NEDC-34041P, “BWRX-300 GNF2 Fuel Assembly Mechanical Design Report” (ML26097A238). The preliminary analyses and acceptance criteria described in NEDC-34041P are generally consistent with the fuel assembly mechanical evaluations discussed in NEDC-33270P and the methodology described in NEDE-21175-3-P-A “BWR Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loading (Amendment No. 3)” (ML102290143). NEDE-21175-3-P-A provides the NRC-approved analysis methodology and acceptance criteria for BWR/4-5 and BWR/6 fuel assembly components under combined LOCA and SSE acceleration loads. As will be discussed in the subsection titled “Design Evaluation of Fuel Assembly Structural Response” in Section 4.2.3.3, of this report, NEDC-34041P does not use plant-specific or site-specific loadings, which was deferred to the FSAR.

The preliminary thermal mechanical operating limits for all GNF2 fuel, including BWRX-300, are defined by power-exposure envelopes for each fuel type that define allowable peak linear heat generation rate and peak burnup. The staff reviewed these preliminary limits and their technical bases, provided in NEDC-34042P.

PRIME analyses are performed in each area to provide an upper bound estimate of relevant figures of merit using either a worst-case tolerance stack-up or a statistically bounding tolerance. The methodology in NEDC-33256P-A, Revision 1 and NEDC-33840P-A specifies that for each analysis, operating conditions are assumed that bound the anticipated conditions of normal operations and AOOs.

#### *4.2.3.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.2.1.2 of this report also applies to this subsection.

#### *4.2.3.3 Technical Evaluation*

The staff reviewed the fuel design for the BWRX-300. The staff followed the guidance in SRP Section 4.2 to ensure that (1) the fuel system is not damaged during normal operations and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) core coolability is always maintained.

PSAR Section 4.2 references NEDC-34042P and NEDC-34041P. These documents provide the technical basis and key information underlying the summary information within the PSAR pertaining to fuel assembly mechanical and fuel rod thermal-mechanical performance. The staff reviewed the technical contents of these documents together with the PSAR to provide a technical evaluation of the performance of GNF2 fuel assembly in the BWRX-300 reactor. The review focused on ensuring that the applicant executed the following:

specified appropriate SAFDLs;

specified and used approved codes and methods and remained within the range of applicability for these codes and methods;

obtained preliminary results that demonstrate that the SAFDLs will not be exceeded for Type 1 AOOs; and

obtained preliminary results that demonstrate that the fuel seismic and dynamic design limits will not be exceeded.

<u>Design Bases: Fuel System Damage</u>

Fuel system damage criteria should ensure that fuel system dimensions remain within their operational tolerances and that the fuel will function as assumed in the safety analyses. The sections below address the following fuel system damage criteria that are listed in SRP Section 4.2 to meet the requirements specified in GDC 10 as it relates to SAFDLs for normal operation, including type 1 AOOs:

stress/strain limits

fuel assembly component fatigue

fuel rod fretting

oxidation and hydriding

dimensional changes (bowing/growth)

rod internal pressure

fuel assembly liftoff

control rod insertability

*Stress/Strain Limits*

Limits are defined and analyzed in three distinct areas: cladding strain, cladding stress, and assembly component stresses. Each area is discussed below.

Cladding Strain

PSAR Section 4.2.3.5 references the PRIME cladding strain limit and method during rapid power ramp Type 1 AOOs. This SAFDL has been previously approved and is therefore acceptable for the BWRX-300. There is no limit on total cladding strain from steady-state irradiation. This approach has also previously been approved and is therefore acceptable for the BWRX-300.

Section 4.1.1 of NEDC-34042P describes the methodology that will be used to perform the design analysis for cladding strain. The PRIME strain methodology described in NEDC-33258P-Ais applied. This preliminary analysis uses a worst tolerance analysis to assure that cladding hoop strain during power ramps will not exceed the stated limit. The staff reviewed the methodology, list of parameters varied in this analysis, and the results of the preliminary cladding strain analysis along the allowable power operating envelope versus peak burnup for each fuel rod type present in the GNF2 fuel bundle. The staff found the preliminary strain analyses to be generally consistent with the PRIME methodology.

Based on this information, the staff concludes that the use of a previously approved transient cladding strain limit and method supports a preliminary determination that the GNF2 fuel design can meet the specified limits on cladding strain when operated within the preliminary power-exposure envelopes for the BWRX-300 in accordance with the guidance in SRP Section 4.2, and is sufficient to support issuance of a CP, with a final review to be performed during the OL review.

Cladding Stress

PSAR Section 4.2.3.10 states that the cladding stress analysis is described in NEDC-33270P. NEDC-33270P describes the design criterion for cladding stress analysis beyond the stress on the cladding from the difference between the rod internal pressure and the coolant system pressure. For these analyses, the effective stress shall be less than the yield stress. This approach has previously been approved and is therefore acceptable for the BWRX-300.

NEDC-33270P describes the methodology that will be used to perform the design analysis for cladding stress, however as stated in PSAR Section 4.2.3.10, the PRIME code will be used instead of the older GESTR code to generate inputs for stress analysis. This analysis uses a square root sum of squares to assure with high confidence that effective stress calculated will not exceed the yield stress. The staff reviewed the methodology and list of parameters varied in this analysis and the results are sufficient for the PSAR to support issuance of a CP.

The NRC staff reviewed the preliminary results of the cladding stress analysis at 100 percent and 130 percent rated power at various times throughout the rod life in NEDC-34042P. The maximum design ratio (ratio of predicted stress to yield stress) was shown to occur at beginning of life for the limiting rod types with significant margin to the limit. The staff confirmed that the results of the applicant's preliminary analysis demonstrate that the power exposure envelopes for the GNF2 fuel design were generally as expected.

Based on this information, the staff concludes that the applicant included sufficient information on a preliminary basis for the purpose of issuing a CP, to demonstrate that the GNF2 fuel design can meet the specified cladding stress limits when operated within the preliminary power-exposure envelopes in accordance with the guidance in SRP Section 4.2.

Assembly Component Stresses

PSAR Section 4.2.5, "Safety Evaluation," references NEDC-33270P and NEDC-33041P which addresses fuel assembly component performance with respect to applied external loads. Section 3.2.1 "Stress, Strain, and Fatigue" of NEDC-33270P provides the design loads and design criteria for assembly component stresses. The applicant stated during the regulatory audit that, because these are based on the loads from steady-state operation, anticipated transients and accident loads, the final loads will only be complete at the FSAR stage. Therefore, the staff performed a preliminary assessment for the purpose of the CP review. The staff confirmed that the design loads are as described in the GESTAR II design methodology (NEDE-24011-P-A) and its supporting references (NEDE-21354-2-P, NEDE-23542-P, and NEDE-21175-3-P-A), based on preliminary information. The design limits/criteria were taken from ANSI/ANS-57.5-1981, which defines a design ratio of effective stress to the material ultimate strength per the approved methodology in NEDE-24011-P-A. Section 3.2.1 of NEDC-33270 also provides assembly component stress evaluations with respect to the design loads. The staff also reviewed NEDC-34041P and audited the engineering analysis reports supporting NEDC-34041P. The analyses of the upper tie plate, lower tie plate, and fuel channel were performed using finite element analysis. A detailed description of the finite element models was not available for review, but sufficient detail was provided for the staff to conclude that the geometric simplifications, element types, and boundary conditions were reasonable for the purposes of a preliminary assessment of the analysis to authorize construction. It is expected that the staff will perform a more detailed review of the final finite element analyses, including confirmation that the actual stresses would not exceed the design limits in their review of the OLA.

Section 3.2.1 "Stress, Strain, and Fatigue" of NEDC-33270P provides the design loads and preliminary stress evaluations for the upper tie plate. The limiting design loads for the upper tie plate result from bundle handling. The analysis was conducted via finite element analysis. A bounding design load of three times the bundle weight was applied in the analysis, consistent with the bounding load in NEDE-23542-P (NEDE-23542 is referenced in GESTAR II). Sufficient detail was provided for the staff to conclude that the analysis methods, material properties, and applied loads/constraints were appropriate for the analysis. Staff concluded that the effective stresses in the grid portion of the upper tie plate were within design limits. The finite element analysis predicted effective stress in the upper tie plate handle exceeded the yield strength of the upper tie plate material at the limiting temperature. The applicant conducted mechanical testing on the upper tie plate to confirm that excessive deformation does not occur under the design loads. Mechanical testing confirmed that the upper tie plate will not experience excessive deformation during bundle handling.

Section 3.2.1 “Stress, Strain, and Fatigue” of NEDC-33270P provides the design loads and preliminary stress evaluation results for the lower tie plate. The limiting design loads for the lower tie plate result from bundle handling. The analysis was conducted via finite element analysis. Design loads, consistent with the bounding loads in NEDE-23542-P, were considered in the analysis (NEDE-23542-P is referenced in version of GESTAR II). Sufficient detail was provided for the staff to conclude that the analysis methods, material properties, and applied loads/constraints were appropriate for the preliminary analysis. Staff concluded that the preliminary effective stresses in the lower tie plate were within design limits defined by the yield strength of the lower tie plate material at the limiting temperature. In the review of NEDC-34041P, the staff observed that the analysis was slightly different than that performed in NEDC-33270P. The finite element analysis model was updated and performs the analysis using the defender debris filter lower tie plate. NEDC-33270P describes that the defender debris filter lower tie plate is an optional feature of GNF2. The NEDC-33270P lower tie plate stress analysis was performed using the debris shield lower tie plate, which is stated as being standard for the GNF2 assembly. Despite the differences, the calculation in NEDC-34041P reaches the same conclusion for the lower tie plate as that in NEDC-33270P, that the stresses are well below the yield stress. The NRC will further evaluate the BWRX-300 specific GNF2 stress analysis in the OL review.

Section 3.2.1 “Stress, Strain, and Fatigue” of NEDC-33270P provides the design loads and preliminary stress evaluation for the fuel channel. The limiting design loads for the channel result from the operating differential pressure defined in NEDE-21354-2-P (referenced in GESTAR II). The preliminary analysis was conducted via finite element analysis. Sufficient detail was provided for the staff to conclude that the analysis methods, material properties, and applied loads/constraints were appropriate for the preliminary analysis. Staff concluded that the effective stresses in the fuel channel were within design limits for the preliminary analysis. The staff noted that small margin exists with respect to fuel channel stresses under the design basis pressure gradient, but that for the analysis of buckling under lateral seismic dynamic loads, it is conservative to neglect the pressure gradient. Seismic dynamic loads acting on the fuel channel are addressed in the section on “Evaluation of the Fuel Assembly’s Structural Response to Externally Applied Forces” of this SER. Channel fatigue is addressed in “Fuel Assembly Component Fatigue” in this section.

The NRC staff reviewed the preliminary stress evaluations of the channel fastener bolt, plenum spring, expansion spring, and the channel fastener spring in NEDC-34041P. The NRC staff found the preliminary stress evaluations reasonable but will evaluate the final stress evaluation during the OL review.

Section 3.2.1 “Stress, Strain, and Fatigue” of NEDC-33270P provides the design loads and stress evaluation of the water rods under differential pressure. The limiting design loads result from differential pressure and operating loads. The preliminary analysis was conducted via analytical solutions. Sufficient detail was provided for the staff to conclude that the analysis methods, material properties, and applied loads/constraints were appropriate for the analysis. Staff concluded that the preliminary effective stress in the water rods under differential pressure was within design limits.

Spacer strength is addressed in the section on “Evaluation of the Fuel Assembly’s Structural Response to Externally Applied Forces” of this SER.

Based on this information, the staff concludes that sufficient information was provided relative to the fuel assembly component stresses and their applicable design limits to authorize construction.

*Fuel Assembly Component Fatigue*

PSAR Section 4.2.3.8 states that the design analysis for cladding fatigue will be performed using PRIME. The design criterion for cladding fatigue stated in the PRIME method has been previously reviewed and approved in the PRIME Topical Report (NEDC-33258P-A). The design criterion is that the fuel rod cladding fatigue life usage shall not exceed the material fatigue capability. The PRIME methodology defines fatigue cycles and uses the square root sum of squares to assure with high confidence that cladding fatigue usage factor will not exceed 1.0. During the staff review of NEDC-34042P, the NRC staff observed that the fatigue analysis for BWRX-300 GNF2 fuel was modified to account for daily load following. Specifically, the number of load cycles increased in the fatigue analysis. As noted in Section 4.2.3.1 of this report, the staff considers that daily load follow operation may impact other aspects of the PRIME methodology, beyond the cladding fatigue analysis. Therefore, the staff is deferring its review of daily load follow to the OL phase once information specified in Section 4.2.3.1 of this report is provided by the applicant at the OL stage. For the purpose of issuing a CP, NRC staff finds the fatigue analysis methodology and criteria in the PRIME method to be reasonable for the PSAR because the NRC is not approving load follow operation in the BWRX-300 in the PSAR review.

Section 3.2.3.2, “Metal Thinning Effects on Zircaloy Channels” of NEDC-33270P discusses the fatigue analysis of the fuel channel. The design limits were taken as the fatigue damage and stress rupture summation, which is less than 1.0 for satisfactory performance. The staff reviewed the channel fatigue analysis provided in NEDC-34041P for NSF channels. The analysis was presented at a high level, but similar to the analysis presented in NEDC-33270P for zircaloy channels. While only zircaloy channels were analyzed in NEDC-33270P, NSF fatigue was assessed in NEDC-33798P-A, Revision 1, “Application of NSF to GNF Fuel Channel Designs” (ADAMS Package Accession No. ML15273A009). The staff finds that the generic analysis in NEDC-33270P supports a preliminary conclusion that the calculated channel fatigue performance with respect to fact that the basis conditions were within the design limits. The staff also notes that the fatigue damage and stress rupture damage summation are only met for a specified pressure differential, which will need to be confirmed during review of the FSAR.

Based on this information, the staff concludes that the applicant provided sufficient information to support issuance of a CP relative to the specified limits on channel fatigue in accordance with the guidance in SRP Section 4.2.

*Fuel Fretting*

PSAR Section 4.2.3.12 “Fretting Wear Testing” states that fretting tests have been performed on the GNF2 fuel assembly that show adequate fretting resistance. The method used to demonstrate the adequacy of the fuel assembly from a flow induced vibration (FIV) perspective was to compare the GNF2 fuel assembly FIV response to the GE14 fuel assembly FIV response. This analysis is included in NEDC-33270P. The staff confirmed that the fretting tests performed on the GNF2 fuel assemblies were applicable to the preliminary BWRX-300 coolant conditions.

Based on this information, the staff concludes that the applicant demonstrated on a preliminary basis that the GNF2 fuel design can be shown to meet the specified limits on fretting and flow-induced vibrations in accordance with the guidance in SRP Section 4.2, and that this is sufficient to support issuance of a CP.

*Oxidation and Hydriding*

PSAR Section 4.2 and NEDC-33270P do not describe an analysis for corrosion and hydriding of the fuel cladding. The staff reviewed NEDC-34042P, which described a design criterion for cladding oxidation and end-of-life cladding hydrogen content. The NRC staff confirmed that the cladding oxide design criterion is consistent with the oxide design criterion discussed in the Section 3.3.1 of the NRC's SE of the PRIME methodology (ML102600259). The NRC staff also confirmed that the cladding hydrogen concentration limit is equivalent to that described in the PRIME methodology. The staff finds these SAFDLs to be acceptable for the BWRX-300 to support issuance of a CP based on the alignment with the PRIME methodology design limits previously reviewed by the NRC.

The PRIME code does not have a corrosion model but rather assumes linear buildup of oxide as a function of time. Although an upper bound of this linear rate indicates that at the expected maximum BWRX-300 fuel residence time the cladding oxide thickness will not exceed the SAFDL limit, there is no analysis or information that would indicate the GNF2 fuel assembly in the BWRX-300 reactor would behave this way. The applicant did not describe an explicit methodology to confirm the cladding hydrogen content or oxide thickness will not exceed the stated limits but instead relies on historic data to ensure that these limits are not exceeded. The staff agrees that the preliminary environmental conditions of the BWRX-300 reactor (coolant temperature, coolant pressure, and coolant chemistry) are expected to be within the bounds of historic BWR data and therefore the approach of relying on historic data and residence time and peak burnup limits is acceptable to demonstrate compliance with oxide and hydrogen limits. However, corrosion and hydrogen pickup are highly sensitive to plant operating conditions, so the staff will review these aspects at the OL phase. Specifically, the applicant should describe in the FSAR the surveillance program for cladding corrosion to confirm oxide thickness and hydrogen pickup are within expected ranges.

If load follow operation is pursued in the OL, cladding corrosion and hydrogen pickup may be impacted, so the reliance on historic data for cladding corrosion and hydrogen pickup may need to be evaluated further in the review of the FSAR.

Based on this information and the use of the NRC-approved PRIME fuel performance methodology, the staff concludes that the preliminary information supplied by the applicant is sufficient to authorize construction relative to the specified limits on cladding oxidation and hydrogen content when operated within the preliminary BWRX-300 power-exposure envelopes, and providing that the information necessary to support the OL review is provided.

NEDC-33270P evaluates GNF2 zircaloy channel corrosion. The BWRX-300 employs NSF channels, which have demonstrated improved corrosion performance compared to zircaloy channels, as evaluated in NEDE-33798P-A, Revision 1. Since NSF displays superior corrosion performance to zircaloy channels, the NRC staff finds the NEDC-33270P channel corrosion evaluation to be acceptable to authorize construction.

Based on this information, the staff concludes that the preliminary information supplied by the applicant is sufficient to support issuance of a CP relative to the specified limits on corrosion and hydrogen content for the BWRX-300 reactor in accordance with the guidance in SRP Section 4.2.

*Dimensional Changes*

The staff reviewed the preliminary evaluations of the GNF2 fuel assembly's capability to accommodate dimensional changes presented in NEDC-34041P. Preliminary evaluations were provided for various assembly components, including evaluations of the fuel rod and water rod upper end plug engagement lengths in the upper tie plate, the fuel channel overlap with the lower tie plate, the channel fastener spring capability to accommodate differing axial lengths of adjacent fuel assemblies, and channel bulge and bow. Similarly, NEDE-33798P-A, Revision 1 evaluates the NSF channel bulge and bow. NEDE-33798P, "NSF Channel Annual Experience Summary Report" Supplement 1, Revision 7 (ML24012A008) provides additional NSF channel operating experience data that demonstrates that NSF channel has superior bow and bulge performance compared to zircaloy channels. The operating experience with the GNF2 fuel assembly and the NSF channel provide reasonable assurance that the BWRX-300 fuel assembly and channel can accommodate dimensional changes on a preliminary basis to support issuance of the CP.

*Rod Internal Pressure*

Section 4.2.3.6 of the PSAR states that the rod internal pressure analysis is performed with the PRIME methodology described in NEDC-33258P-A. The PRIME methodology specifies that the design criterion for rod internal pressure is that the rod internal pressure shall be less than the critical pressure, where the critical pressure is determined as the pressure where the cladding creep out rate is equal to the fuel swelling rate. This SAFDL has been previously approved in the PRIME methodology and is thus acceptable for the BWRX-300.

The NRC staff reviewed the results of the NEDC-34042P BWRX-300 rod internal pressure design analysis along the allowable power operating envelope versus peak burnup for each fuel rod type present in the GNF2 fuel bundle. The staff confirmed that the results of the applicant's analysis show that the power exposure envelopes for the GNF2 fuel design are reasonable because they preliminarily ensure that cladding liftoff will not occur, consistent with the PRIME methodology.

Based on this information, the staff concludes that the applicant demonstrated that the GNF2 fuel design meets the specified limits on rod internal pressure when operated within the preliminary BWRX-300 power-exposure envelopes provided in in accordance with the guidance in SRP Section 4.2.

*Fuel Assembly Liftoff*

PSAR Section 4.2.3.4 discusses fuel assembly liftoff from hydraulic pressure, seismic and dynamic loads. The analyses of seismic and dynamic loads are discussed in this SER in the following section on "Evaluation of the Fuel Assembly's Structural Response to Externally Applied Forces." The lift analysis requires site specific seismic parameters and must therefore be provided as a part of the plant evaluation with the FSAR (see the section titled "Design Evaluation of Fuel Assembly Structural Response" later in this section for the related OL action item).

The staff concludes that the applicant demonstrated that the GNF2 fuel design in the BWRX-300 reactor provides sufficient information relative to the specified limits for fuel assembly liftoff in accordance with the guidance in SRP Section 4.2, pending a site-specific analysis described above at the OL stage.

*Control Rod Insertability*

PSAR Section 4.2.3.4, "Fuel Lift and Seismic and Dynamic Load Analysis," addresses fuel liftoff and unseating of the fuel assembly, which could prevent control rod insertability. The section on "Evaluation of the Fuel Assembly's Structural Response to Externally Applied Forces" of this SER provides the staff's evaluation of control rod insertability under seismic and loss-of-coolant accident (LOCA) loads.

<u>Design Bases: Fuel Rod Failure</u>

Fuel rod failure should not occur as a result of specific causes during normal operation and AOOs, but it is permitted as a result of postulated accidents. The sections below address the following fuel system damage criteria that are listed in SRP Section 4.2.

hydriding

cladding collapse

overheating of the cladding

overheating of the fuel pellet

excessive fuel enthalpy

pellet/cladding interaction

bursting

mechanical fracturing

Some of these limits apply to postulated accidents and are primarily addressed in Chapter 15 of the PSAR. The staff's evaluation of these will be addressed in Chapter 15 of the SER as noted for the specific mechanisms below.

*Hydriding*

Section 3.2.4 of NEDC-33270P describes the design criterion for rod internal sources of hydrogen This SAFDL is acceptable for the BWRX-300 because it is consistent with GESTAR II and it ensures that hydrogenous impurities are limited through these specifications such that the impurities will not cause rod failure due to localized primary hydriding.

The staff reviewed NEDC-34042P, which describes successful operating and manufacturing experience designing to the stated fabrication limit. The operating experience with these fuel rods demonstrates that hydride-induced embrittlement is not an active failure mechanism for current GVH fuel designs, including GNF2.

The staff concludes that the applicant provided sufficient information to authorize construction relative to the specified limits with respect to internal hydriding in accordance with the guidance in SRP Section 4.2.

*Cladding Collapse*

PSAR Section 4.2.3.9 references that the cladding creep collapse methodology is described in the previously approved NEDC-33139P-A, "Cladding Creep Collapse" (ADAMS Accession No. ML14094A228). The staff finds this methodology acceptable on a preliminary basis for the GNF2 fuel assembly in the BWRX-300 reactor as none of the expected preliminary values for the relevant reactor parameters exceed the bounds of approval for this methodology.

Based on this information, the staff concludes that the applicant provided sufficient information relative to the specified limits on cladding collapse to authorize construction, when operated within the power-exposure envelopes provided in NEDC-34042P in accordance with the guidance in SRP Section 4.2.

*Overheating of the Cladding*

The guidance in SRP Section 4.2 states that failures are assumed to be precluded if the thermal-margin criteria (critical power ratio [CPR] for BWRs) are satisfied. Section 4.4 of this report discusses the CPR margin analysis for normal operation and Type 1 AOOs (Type 1 AOOs are defined in Chapter 15.1.3.1 of this report). The NRC staff find the use of CPR acceptable because it meets the SRP Section 4.2 guidance.

The various design-basis event evaluations, as detailed in Chapter 15 of this report, document the cladding temperature under postulated accident conditions, which deviate from the SRP Section 4.2 guidance for assuming failure using CPR. The NRC staff's evaluation of this deviation from the SRP Section 4.2 guidance for Type 2 AOOs (Type 2 AOOs are defined in Chapter 15.1.3.1 of this report) and design-basis accidents (DBAs) is documented in Chapter 15 of this report (e.g., Section 15.1.3.4).

*Overheating of Fuel Pellets*

NEDC-33270P describes the design criterion for fuel temperature, which is that the maximum fuel temperature shall be below the melting temperature. This SAFDL has been previously approved by the NRC in the PRIME methodology and in GESTAR II and is therefore acceptable for the BWRX-300.

The NRC reviewed the preliminary fuel pellet temperature analysis for BWRX-300 GNF2 fuel in NEDC-34042P and confirmed that the preliminary methodology was consistent with that approved in the PRIME methodology and the fuel pellet temperature calculation would not be invalidated by the BWRX-300 reactor differences.

The preliminary calculations in NEDC-34042P show the results of the fuel maximum temperature analysis along the preliminary power operating envelope versus peak burnup for each fuel rod type present in the GNF2 fuel bundle. The staff confirmed that the results of the applicant's preliminary analysis demonstrate that the power exposure envelopes for the GNF2 fuel design preclude fuel melt.

Based on this information, the staff concludes that sufficient information was provided relative to the specified limits on maximum fuel temperature to authorize construction, when operated within the preliminary power-exposure envelopes in accordance with the guidance in SRP Section 4.2.

*Excessive Fuel Enthalpy*

The staff's evaluation in Chapter 15 of this report (e.g., Section 15.1.3.4) documents the review of a sudden increase in fuel enthalpy from a reactivity-initiated accident below the fuel melting temperature.

*Pellet/Cladding Interaction*

Start-up and rapid power ramps causing a failure by pellet/cladding interaction (PCI) due to a combination of mechanical and chemical interactions. In order to eliminate this failure mechanism, GVH introduced liner cladding which has an inner liner of natural zirconium. This liner is in the GNF2 fuel assembly design, as described in NEDC-33270P and has proven effective in reducing instances of PCI failure. GVH has extensive experience with GNF2 assemblies that contain liner cladding, demonstrating minimal failure rates.

The staff finds PCI is adequately addressed and is acceptable at the PSAR stage to support issuance of the CP because the GNF2 operating experience to date demonstrates acceptable performance with respect to PCI.

*Bursting*

Chapter 15 of this report (e.g., Section 15.1.3.4) presents the staff's evaluation of fuel rod bursting.

*Mechanical Fracturing*

The following section on "Evaluation of the Fuel Assembly's Structural Response to Externally Applied Forces," presents the staff's evaluation of fuel rod mechanical fracturing.

<u>Design Bases: Fuel Coolability</u>

Damage to fuel rods and fuel assemblies should never be so severe as to result in a loss of coolable geometry. The limits below are listed in SRP Section 4.2.

cladding embrittlement

violent expulsion of fuel

generalized cladding melting

fuel rod ballooning

structural deformation

These limits apply to postulated accidents and are primarily addressed in Chapter 15 of the PSAR. The following section on "Evaluation of the Fuel Assembly's Structural Response to Externally Applied Forces" gives the staff's evaluation of fuel assembly distortion, mechanical fracturing, and fuel assembly structural damage from external forces, related to the fuel assembly's structural response to externally applied loads.

*Evaluation of the Fuel Assembly's Structural Response to Externally Applied Forces*

This section provides a technical evaluation of the fuel assembly's structural response to externally applied forces. The first section below discusses the design requirements and acceptance criteria. The second section provides a review of the design evaluation that was provided by the applicant. A detailed description of the finite element models was not available for review at the PSAR stage, but sufficient detail was provided for the staff to conclude that the geometric simplifications, element types, and boundary conditions were reasonable. The staff expects to perform a more detailed review of the finite element analyses in the review of the OL.

Design Requirements and Acceptance Criteria for Fuel Assembly Structural Response

PSAR Section 4.2.3.4, "Fuel Lift and Seismic and Dynamic Load Analysis" provides reference to NEDC-33270P, Revision 11, which describes GNF2 compliance with the specified limits defined in GESTAR II (NEDE-24011-P-A). PSAR Section 4.2.3.4 also provides reference to NEDE-21175-3-P-A. NEDE-21175-3-P-A provides the NRC-approved analysis methodology and acceptance criteria for BWR/4-5 and BWR/6 fuel assembly components under combined LOCA and SSE acceleration loads. The acceptance criteria are that the calculated effective stress resulting from combined LOCA and SSE loads are limited to a fraction of the material ultimate strength. The faulted condition acceptance criteria do not necessarily preclude permanent deformation, as they are based on a fraction of the ultimate tensile strength, not the yield strength. Section 3.2.1 of NEDC-33270P states that the channel stresses are well below the failure strength at operating conditions. Moreover, it is explained in Section 3.2.7 of NEDC-33270P that the channel buckling analysis under faulted loads is performed per the approved methodology and results in the same margin as in NEDE-21175-3-P-A, which ensures control rods can be inserted. Therefore, the acceptance criteria are reasonable for the PSAR to support issuance of the CP but will be further evaluated in the review of the OL.

The combined LOCA and SSE loads defined in NEDE-21175-3-P-A are defined as a fuel assembly capability envelope of horizontal and vertical accelerations that are not plant-specific but are intended to be bounding. Rather, the methodology in NEDE-21175-3-P-A states that the fuel assembly design will have satisfied the acceptance criteria if plant-specific horizontal and vertical accelerations fall within the capability envelope. The applicant stated in NEDC-33270P that the GNF2 fuel design, when modeled, demonstrates similar dynamic behavior to the fuel design considered in NEDE-21175-3-P-A, which validates the use of the methodology and therefore the NRC staff finds the method to be acceptable for the PSAR to support issuance of the CP.

NEDE-21175-3-P-A, Revision 0 defines a maximum horizontal acceleration (depending on channel type) and a maximum vertical acceleration capability envelope. During the audit, the applicant stated that the GNF2 analyses are based on design basis horizontal and vertical accelerations that correspond to approximately the design basis limits for the fuel design. The NRC will evaluate the validity of the accelerations assumed in the review of the FSAR. The fuel rod and water rod acceptance criteria were evaluated in accordance with the acceptance criteria described in NEDE-21175-3-P-A, Revision 0. The acceptance criteria for the spacers, channel, lower tie plate, and upper tie plate were established through mechanical testing.

Based on this information, the staff concludes that the applicant demonstrated that the GNF2 fuel seismic/dynamic analysis has reasonable acceptance criteria, design requirements, and methods suitable for the PSAR to support issuance of the CP, however, more plant specific details will be needed in the FSAR (see the related discussion in the section titled "Design Evaluation of Fuel Assembly Structural Response," below).

Design Evaluation of Fuel Assembly Structural Response

The NRC staff reviewed the fuel mechanical evaluation for external loads in NEDC-34041P. NEDC-34041P describes the preliminary model development, design loads, acceptance criteria, and analytical results for the GNF2 design evaluation. The applicant stated that the design evaluations were in accordance with the approved methodology described in NEDE-21175-3-P-A, Revision 0, using conditions intended to be bounding. The NRC will review the analysis conditions and the plant-specific conditions in the FSAR. NEDC-34041P notes that the plant-specific applicability is outside the scope of the document. Plant specific applicability information was not provided with the PSAR and must be submitted with the FSAR. The following OL action item is specified in this SER:

For the OLA, the FSAR must provide the design evaluation of fuel assembly structural response and the hydraulic lift analysis using plant-specific seismic parameters, or a bounding analysis must be provided with a demonstration that the analysis bounds the plant-specific seismic conditions

During the audit, the applicant stated that the preliminary GNF2 mechanical analyses in NEDC-34041P are intended to be bounding on design basis horizontal and vertical accelerations that the applicant has stated are approximately the design basis limits for fuel design. The calculated stresses and loads were then compared to the corresponding faulted design criteria. From these evaluations using bounding design loads, the applicant stated that fuel assembly structural integrity and bundle coolability is assured and that mechanical fragmentation is precluded, provided that the analysis bounds site-specific conditions.

PSAR Section 4.2.3.4, "Fuel Lift and Seismic and Dynamic Load Analysis," addresses control rod insertability. Further technical details are provided in Section 3.2.7 "Control Rod Insertion," of NEDC-33270P, Revision 11, which describes the evaluation of fuel assembly liftoff that would prevent control rod insertion, and the applicability of the previously approved analysis presented in NEDE-21175-3-P-A, Revision 0, which demonstrated significant margin to assembly unseating. NEDC-33270P, Revision 11 and NEDC-34041, Revision 0 describe the component stress and buckling evaluations that provide assurance that fuel channel deformation does not affect control rod insertion. Section 3.2.5 of NEDC-33270P, Revision 11 addresses fuel rod and water rod buckling and compliance with the design criteria. Additionally, regarding hydraulic lift loads, the PSAR states that the GESTAR II methodology will be used and provides a qualitative statement that the lift load will be lower in BWRX-300 with natural circulation than in a BWR with forced circulation. The NRC staff finds this argument reasonable to support issuance of the CP. However, the lift analysis requires site specific seismic parameters and must therefore be provided as a part of the plant evaluation with the FSAR (see OL action item above).

In the stress evaluation methodology under combined SSE and LOCA loading, the unfaulted condition stress analysis of the fuel channel that the NRC staff reviewed in NEDC-34041, Revision 0 suggests limited margin under the assumed pressure gradient, while the faulted condition analysis neglects this pressure gradient. During the audit, the applicant clarified that neglecting the pressure gradient is bounding as the pressure gradient generates a geometric stiffening effect that increases the maximum allowable bending moment. This is substantiated in the approved methodology of NEDE-21175-3-P-A where it demonstrated via channel bending tests (in Table II.7-1) that the pressure differential tends to increase the failure moment, therefore it is reasonable for the PSAR. The NRC staff will further evaluate the margin in the unfaulted stress analysis of the fuel channel and the assumption of neglecting the pressure gradient in its review of the OL.

Based on the staff's review of the material presented in the PSAR and supporting documents, the staff concludes that the applicant preliminarily analyzed the loads and stresses reasonably and presented reasonable acceptance criteria for the PSAR to support issuance of the CP. The preliminary analysis adequately addressed fuel mechanical fracturing, fuel system distortion, control rod insertability, and coolable geometry under seismic and LOCA loads on a preliminary basis for the PSAR. The NRC staff will further evaluate the margin in the unfaulted stress analysis of the fuel channel and the assumption of neglecting the pressure gradient in the review of the OL as well as the faulted condition acceptance criteria in the OLA review.

**Operating Experience**

The staff reviewed GVH's operating experience with recent 10×10 designs including GNF2 as part of the audit. The operating experience data shows extensive experience of over 100 reloads of GNF2 with a low fuel failure fraction. The staff notes that the GNF2 fuel assembly for BWRX-300 is identical to the GNF2 fuel assemblies that have been irradiated. The staff further notes that the preliminary BWRX-300 plant operational parameters important to fuel behavior are not significantly different from those in the GVH operating fleet. The NRC staff also considered the similar operating experience data presented to the NRC at annual fuel update meetings with GVH (e.g., see ADAMS Package Accession No. ML24312A391). The data reviewed during the audit confirmed the expected trends based on the data observed in NRC–GVH annual fuel update meetings. Therefore, the staff finds that the GVH operating experience applies to GNF2 fuel assemblies in the BWRX-300 reactor on a preliminary basis, sufficient to support issuance of the CP.

*4.2.3.4 Conclusion*

The staff performed a technical review of the fuel assembly design evaluation. The assembly design bases and SAFDLs that are used to demonstrate compliance are the same as those used for the previously approved GNF2. The methodology codes, and application of uncertainties are specified in the previously approved GESTAR II methodology. The primary differences are the plant and reactor-specific inputs, which are considered preliminary at the PSAR stage. Generic GNF2 calculations in NEDC-33270P and the preliminary calculations in NEDC-34041P and NEDC-34042P were reviewed. The analyses are considered sufficient to support issuance of the CP and demonstrate that the BWRX-300 reactor design is capable of ensuring SAFDLs will not be exceeded for conditions of normal operation including the effects of Type 1 AOOs, pending final confirmation during the OL review. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

There were several analyses that were deferred by the applicant to the review of the FSAR, and are reflected in OL action items listed previously in this section. These are:

hydraulic lift analysis

seismic and LOCA fuel assembly structural analyses using plant specific inputs or a demonstration that the analyses would bound the plant-specific analyses

daily load follow

### 4.2.4 Control Rod Design Bases

#### *4.2.4.1 Introduction*

The control rods are cruciform shaped elements, which occupy alternate spaces between fuel assemblies throughout the core. The control rods are connected to bottom vessel mounted drive mechanisms, which allow either axial positioning within the core or withdrawal into the guide tubes below the core. The outer fuel channel surfaces of the four fuel assemblies in a cell guide the insertion and withdrawal of the control rod. The control rods are cooled by the core leakage (bypass) flow and CRD purge flow to remove heat generated by neutron and gamma absorptions.

#### *4.2.4.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.2.1.2 of this report also applies to this subsection.

#### *4.2.4.3 Technical Evaluation*

PSAR Section 4.2.4.2 only describes the control rod design criteria at a high level. The applicant stated that the design of the BWRX-300 “Ultra-Plus”™ control rod is in progress and provided an in-progress description of the design specifications. The high-level control rod design criteria address structural material, reactivity, and insertability considerations. The preliminary criteria are sufficient to support issuance of the CP because, provided those criteria are met, the control rod design would be capable of reliably controlling reactivity and shutting the reactor down during normal operation and postulated accidents. The applicant further stated that an evaluation demonstrating that the acceptance criteria are met will be provided in the FSAR. Therefore, at the OL phase, the applicant should provide control rod design acceptance criteria and safety analyses in the FSAR.

No technical evaluation of the control rod design bases was performed.

#### *4.2.4.4 Conclusion*

The staff concluded that the preliminary control rod high level design bases described in PSAR Section 4.2.4.2 are sufficient to support issuance of the CP. However, a detailed evaluation of the control rod design was not performed because the design is in-progress; therefore, the staff will review control rod design and safety analysis during the review of the OL and the FSAR.

### 4.2.5 Safety Evaluation

#### *4.2.5.1 Introduction*

As previously stated, PRIME analyses are performed in each area to provide an upper bound estimate of relevant figures of merit using either a worst-case tolerance stack-up or a statistically bounding tolerance. For each analysis operating conditions are assumed that bound the anticipated conditions of normal operations and Type 1 AOOs.

The applicant has provided a number of documents that show preliminary safety analysis results to assist the staff in their review. The preliminary fuel rod thermal-mechanical results are provided in NEDC-33270P for the generic GNF2 fuel assembly and in NEDC-34041P and NEDC-34042P specifically for the GNF2 fuel assembly in the BWRX-300 reactor.

#### *4.2.5.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.2.1.2 also applies to this subsection.

#### *4.2.5.3 Technical Evaluation*

The NRC staff evaluated each of the analyses used to estimate the relevant figure of merit related to fuel thermal-mechanical performance in prior sections of this SER and found them to be sufficient to support issuance of the CP.

#### *4.2.5.4 Conclusion*

The NRC staff found that the GNF2 fuel thermal-mechanical analyses in BWRX-300 are sufficient to support issuance of the CP, based on the evaluations in previous subsections of this final safety evaluation report (FSER). Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

### 4.2.6 Testing, Inspection and Surveillance Plans

#### *4.2.6.1 Introduction*

PSAR Section 4.2.6 “Testing, Inspection and Surveillance” provides reference to NEDE-24011-P-A-31, “General Electric Standard Application for Reactor Fuel,” which documents the GVH program for surveillance of production and developmental fuel.

#### *4.2.6.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.2.1.2 of this report also applies to this subsection.

#### *4.2.6.3 Technical Evaluation*

The GVH program for surveillance program for production and developmental fuel has been previously approved and applied to the GNF2 fuel assembly in operating reactors to date. However, as discussed in Section 4.2.3 of this report, since cladding corrosion is highly dependent on plant-specific conditions and the applicant is relying on historical data for their safety case, the staff will review the applicant’s surveillance program for cladding oxide thickness and hydrogen content at the OL licensing phase.

Based on this information, the staff concludes that the applicant demonstrated on a preliminary basis that the GNF2 fuel design is acceptable to support issuance of the CP.

Additionally, if load follow operation is pursued in the OL, the NRC staff will further evaluate whether the GNF testing, inspection, and surveillance plans described in the FSAR are adequate.

#### *4.2.6.4 Conclusion*

The staff concludes that the GVH program for production and developmental fuel is acceptable because it employs the previously approved surveillance program in NEDE-24011-P-A-31. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

### 4.2.7 Conclusion

With the exception of those items identified for review in the FSAR listed below, or otherwise noted in the previous sections of this report, the staff concludes preliminarily that the fuel system of the BWRX-300 has been designed so that it can be demonstrated at the OL stage to meet the following requirements: (1) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (2) fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained, even after severe postulated accidents, relative to the related requirements of 10 CFR 50.46; GDC 10, 27, and 35 in Appendix A to 10 CFR Part 50; and 10 CFR 50.34. The staff concludes that its review of the information provided in the PSAR is sufficient to support the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a) and 10 CFR 50.35, as applicable.

As described in the staff's safety evaluation above, the FSAR submitted with the OLA should include additional technical information to address the items noted previously in this section.

## 4.3 <u>Nuclear Design</u>

PSAR Section 4.3 describes elements of the nuclear design bases, analysis techniques, and results of sample analyses, and relates these to regulatory requirements.

In review of the CRN-1 nuclear design, NRC staff considered information contained in the PSAR including supplements, responses to the staff's requests for additional information (RAIs), docketed responses to staff's audit questions, and referenced topical and technical reports. Staff conducted its review in accordance with the guidelines provided by SRP Section 4.3, Revision 3, and the Interim Staff Guidance for Safety Review of Light-Water Power Reactor Construction Permit Applications, DNRL-ISG-2022-01.

### 4.3.1 Nuclear Design Bases

#### *4.3.1.1 Introduction*

PSAR Section 4.3.1 identifies design basis requirements and controlling parameters on the core design.

#### *4.3.1.2 Regulatory Evaluation*

PSAR Table 1.9-4 indicates that PSAR Section 4.3 conforms with the guidance in SRP Section 4.3, "Nuclear Design," of NUREG-0800. NUREG-0800 identifies the following regulatory requirements to be applicable to nuclear design:

10 CFR Part 50, Appendix A GDC 10, “Reactor Design,” as related to the requirement that SAFDLs are not be exceeded during normal operation, including anticipated operational occurrences.

GDC 11, “Reactor Inherent Protection,” as related to the requirement that, in the power operating range, inherent core characteristics provide prompt feedback to compensate for rapid increases in reactivity.

GDC 12, “Suppression of Reactor Power Oscillations,” as related to the requirement that the core be designed such that power oscillations that could result in conditions exceeding SAFDLs are not possible or can be reliably detected and suppressed.

GDC 13, “Instrumentation and Control,” as related to the requirement to provide instrumentation to monitor variables and systems that can affect the fission process over their anticipated ranges during normal operation, for anticipated operational occurrences, and for accident conditions, and to maintain the variables and systems within prescribed operating ranges.

GDC 20, “Protection System Functions,” as related to the requirement that a protection system be provided to ensure automatic initiation reactivity control systems to assure that SAFDLs are not exceeded as a result of AOOs and to ensure automatic operation of systems and components important to safety occurs under accident conditions.

GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” as related to the requirement that that the protection system automatically initiates actions such that SAFDLs are not exceeded for any single malfunction of the reactivity control system.

GDC 26, “Reactivity Control System Redundancy and Capability,” as related to the requirement for at least two independent reactivity control systems of different design, and requirements on the capability to ensure SAFDLs are not exceeded under normal operation and AOOs with appropriate margin for malfunctions such as stuck rods, to control the rate of planned, normal power changes, and to hold the reactor core subcritical under cold conditions. In PSAR Section 3.1.3.7, the applicant identified an alternative PDC 26 to address reactivity control system redundancy.

GDC 27, “Combined Reactivity Control Systems Capability,” as related to the requirement that reactivity control systems, in conjunction with poison addition by the emergency core cooling system, be capable of reliably controlling reactivity changes to assure that the capability to cool the core is maintained under accident conditions.

GDC 28, “Reactivity Limits,” as related to the requirement that reactivity control systems are designed to limit the amount and rate of reactivity increase such that effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary that exceeds limited local yielding nor cause sufficient damage to impair significantly the capability to cool the core. In PSAR Section 3.1.3.9, the applicant identified an alternative PDC 28 to address reactivity limits.

#### *4.3.1.3 Technical Evaluation*

PSAR Section 4.3.1.1 references fuel reactivity acceptance criteria in NEDE-24011-P-A-31-US, “General Electric Standard Application for Reactor Fuel (GESTAR II) (Supplement for United States)” as part of the nuclear design bases. As noted in the NRC staff safety evaluation for GESTAR II, this topical report is applicable to the BWR/2 through BWR/6 plants listed in Table 1-1 of the LTR. NRC staff evaluation of these criteria as applied to the CRN-1 CP is in Section 4.3.4.3 of this report.

PSAR Section 4.3.1.2 discusses design basis requirements for shutdown margin. It specifies that the core will be designed such that it can be 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 coupled hydraulic control unit (HCU) stuck in the full out position with all other control rods fully inserted. It also specifies that the core is assumed to be in a cold, xenon-free condition. The staff considers this treatment of stuck rods in evaluation of shutdown margin to be appropriate for CRN-1, as HCUs provide the safety-related means of inserting control rods and each hydraulic control unit actuates two control rods (with the exception of the central control rod, which has its own hydraulic control unit). PSAR Section 4.3.4.4 clarifies that, for the purposes of the shutdown margin evaluation, cold temperatures from 20 degrees Celsius (°C) to 286°C are evaluated. The staff finds evaluation of a range of temperatures to identify the most reactive state to be an appropriate consideration given that positive values of moderator temperature coefficient have been evaluated for GNF2 lattices below operating temperatures. Ensuring that control rods have sufficient negative reactivity to bring the plant from its most reactive operating state to a cold, xenon-free condition will support the principal design criterion (PDC) 26 requirement to hold the core subcritical under cold conditions. These evaluations will also support that the reactivity control system is designed to supply sufficient negative reactivity to bring the core subcritical with appropriate margin for stuck rods under accident and transient conditions consistent with requirements of PDC 26 and GDC 27. Evaluation of the ability of reactivity control systems to control reactivity changes such that SAFDLs are not exceeded and the capability to cool the core is maintained as needed to meet PDC 26 and GDC 27 is provided in Chapter 15 of this report.

The CRD system is designed with multiple means to insert control rods. In addition to control rods, the boron injection system (BIS), described in PSAR Section 9.3.10, provides an additional means for reactivity control. The role of this system relative to GDC requirements to provide independent means of reactivity control of different design principles is discussed in Section 4.6.

PSAR Section 4.3.1.3.1 identifies that an LHGR operating limit will be defined in order to ensure that fuel operation is maintained within the fuel rod thermal-mechanical and safety analysis bases. It states that the maximum overpower during an AOO is confirmed to not cause fuel melt or excess cladding stress or strain, with further discussion in PSAR Section 4.2.3. PSAR Section 4.2.3 states that a fuel rod thermal-mechanical design analysis is performed using PRIME, as described in NEDC-33256-P-A, NEDC-33257-P-A, NEDC-33258-P-A, and NEDC-33840-P-A. These topical reports, in particular NEDC-33258P-A, "The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance Part 3 – Application," and Section 5.0 of NEDO-33840P-A "The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance," describe the application of PRIME to develop steady-state LHGR limits that ensure SAFDLs (such as limits on cladding strain or fuel temperature) are not exceeded during normal operation or anticipated operational occurrences. The staff understand that these limits will be established to ensure that GDC 10 is met.

Preliminary steady-state LHGR limits were provided in NEDC-34042P (ML26097A239). The NRC staff evaluation of the use of PRIME with the CRN-1 CPA is described in Section 4.2.

Steady-state LHGR limits are dependent on the progression of identified AOO events. NRC review of AOO events identified in the preliminary safety evaluation is discussed in PSAR Chapter 15. The PRIME analyses performed by the applicant in support of the PSAR establish an operating envelope that is intended to cover the expected conditions of normal operation and

AOOs. Because these conditions are not yet finalized, at the FSAR stage, the NRC staff expects to review event-specific implementation of PRIME.

While steady-state limits reflecting the preliminary safety analysis have been provided for audit, the FSAR must establish LHGR operating limits according to the final safety analysis. The response to audit item A-15.2-1 (ML26091A346) identifies that off-rated thermal limits will also be defined in the OLA as required to ensure that full-power conditions remain limiting. LHGR is identified as a future limiting condition for operation in the preliminary technical specification table of contents included in PSAR Table 16.5-1. At the OL licensing stage, the applicant must provide the following additional technical information with the FSAR:

The OLA shall establish limiting conditions for operation for linear heat generation rate pursuant to 10 CFR 50.36. These limiting conditions for operation shall ensure fuel temperature and fuel cladding strain remain within limits needed to avoid fuel rod failure or fuel damage during normal operation, including Type 1 AOOs defined in Section 15.1.3.1 of the staff's safety evaluation report for the CRN-1 CPA. These limiting conditions for operation shall be established according to evaluation models that are approved by NRC staff for application to BWRX-300 and listed in the technical specifications.

The OLA must provide off-rated thermal limits as needed to ensure operation remains within the initial conditions assumed in the safety analysis such that design basis events initiated below rated power meet applicable acceptance criteria.

PSAR Section 4.3.1.3.2 identifies that a minimum critical power ratio (MCPR) operating limit will be established to ensure that the fuel cladding integrity safety limit defined in PSAR Section 4.4.4 will not be exceeded for AOOs, as discussed in PSAR 15.5. The staff review of MCPR is documented in Section 4.4. The applicant confirmed in response to audit item A-15.3-1 (ML26091A346) that PSAR terminology discussing OL MCPR limits is consistent with approved methods, which NRC staff understands account for cycle-specific power distribution and core monitoring uncertainty. That is, OL MCPR is obtained by adding the maximum ΔCPR value for the most limiting AOO to the cycle-specific fuel cladding integrity safety limit, or $MCPR_{99.9\%}$, as specified in PSAR Section 4.4.1.1.3.

Although the CRN-1 CP does not propose technical specifications (consistent with the requirements of 10 CFR 50.34(a)), MCPR is identified as a limiting condition for operation in the preliminary technical specification table of contents included in PSAR Table 16.5-1. As 10 CFR 50.36 requires applicants for licenses authorizing operation of production or utilization facilities to propose technical specifications and the PSAR has identified that a MCPR operating limit will be established to ensure fuel cladding integrity during type 1 AOOs (See Section 15.1.3.1 of this SER for an explanation of type 1 AOOs and type 2 AOOs), the OLA must include the following information:

The OLA shall propose limiting conditions for operation for minimum critical power ratio pursuant to 10 CFR 50.36 that ensure fuel cladding integrity safety limit(s) and cycle-specific fuel cladding integrity safety limits are not violated during normal operation, including Type 1 AOOs defined in Section 15.1.3.1 of the staff's SER for the CRN-1 CPA. These limiting conditions for operation shall be established according to evaluation models that are approved by NRC staff for application to BWRX-300 and listed in the technical specifications.

The OLA shall establish a safety limit for minimum critical power ratio pursuant to 10 CFR 50.36. This safety limit shall be established according to evaluation models that are approved by NRC staff for application to BWRX-300.

The staff notes that PSAR Section 4.3.1.3 discusses operating limits with respect to ensuring SAFDLs are not exceeded pursuant to GDC 10. The applicant has proposed additional criteria to estimate numbers of fuel rod failures in certain event sequences as discussed in Section 15.1.3.5. The set of operating limits provided with the FSAR must also ensure that the plant will be maintained within the initial conditions assumed in these safety analyses.

#### *4.3.1.4 Conclusion*

PSAR Section 4.3.1 identifies design basis requirements and controlling parameters on the core design. The staff finds this information is provided pursuant to 10 CFR 50.34(a)(3)(ii). Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

### 4.3.2 Core Nuclear Design Description

#### *4.3.2.1 Introduction*

PSAR Section 4.3.2 describes basic attributes of the nuclear design of the BWRX-300 reactor core. PSAR Figure 4.3-1 illustrates the layout of fuel assemblies, control rods, and core instrumentation. PSAR Section 4.3.2 also describes an example equilibrium cycle, with placement of fuel assembly batches shown in Figure 4.3-2 and critical control rod positions at sample times-in-cycle shown in Figure 4.3-3. The equilibrium cycle is developed to support preliminary safety analysis. Results from analysis of the equilibrium cycle were also provided. This information is incorporated from NEDC-34044P, “BWRX-300 GNF2 Equilibrium 12 Month Cycle Nuclear Design Report,” and NEDC-34045P, “BWRX-300 GNF2 Fuel Bundle Information Report for Equilibrium 12-Month Cycle”. These reports also include information on lattice design, enrichment, gadolinium loading, loading pattern, and assembly exposure.

#### *4.3.2.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.3.1.2 of this report also applies to this subsection.

#### *4.3.2.3 Technical Evaluation*

The applicant used the equilibrium cycle in the preliminary safety analysis provided in Section 15 of the PSAR. The staff noted that the proposed annual equilibrium cycle is short in comparison to other modern core designs in domestically licensed BWRs. The equilibrium cycle does not include sequence exchanges; the same set of control rods is used to control hot full power reactivity through the end of the cycle.

The applicant’s response to audit item A-4.3-1 (ML26091A346) specifies that an initial cycle design will be provided with the OLA to support the corresponding final safety analysis. Additional discussion of the initial cycle is in Section 4.3.4.3 of this report.

PSAR Section 4.3.2 states that the nuclear instrumentation design depicted in PSAR Figure 4.3-1 is representative. Staff notes that PSAR Figure 7.3-6 also shows the same locations of local power range monitor (LPRM) and gamma thermometer strings and is not labeled as representative. At the OL licensing stage, the staff will review the finalized instrumentation layout provided in the FSAR.

#### 4.3.2.4 Conclusion

PSAR Section 4.3.2, including documents incorporated by reference, describes an equilibrium core design pursuant to 10 CFR 50.34(a)(2) and 10 CFR 50.34(a)(4). Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

### 4.3.3 Core Nuclear Analytical Methods

#### 4.3.3.1 Introduction

PSAR Section 4.3.3 identifies analytical methods used in the steady-state nuclear core analysis for the BWRX-300, i.e., the 2-D lattice physics code TGBLA06 and the 3-D BWR core simulator code PANAC11. PSAR Section 4.3.3 notes that description of these methods, applicability of these methods to CRN-1, and uncertainty quantification for the CRN-1 core design is discussed in NEDC-34039P, “BWRX-300 GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology” (ML26097A242). This technical report references topical reports for TGBLA06 and PANAC11 that were previously approved by the NRC for steady-state core analysis of ESBWR (i.e., NEDC-33239P-A, Revision 5 [ML102800396]) as well as BWR/2 through BWR/6 designs (i.e., NEDO-30130-P-A, Revision 0 and MFN-035-99) and BWR/2 through BWR/6 designs in the EPU/MELLLA+ regime (i.e., NEDC-33173P-A, Revision 5 [ML19228A268]). The staff reviewed this technical report to evaluate applicability of TGBLA06 and PANAC11 to CRN-1 for the purpose of supporting issuance of the CP, as discussed in the following section.

Staff finds nuclear analysis codes, methods and analytical results referenced in PSAR Section 4.3.3 sufficient for the purposes of the CPA. In order to ensure that the licensing basis is clearly established and can be maintained during operation of the facility, the FSAR provided with the OLA should include the following information:

The FSAR must include a listing of the nuclear codes, methods and analytical results (within either technical or topical reports) that are either incorporated by reference, or are described, outlined or summarized in sufficient detail to be considered described within FSAR for the purpose of applying 10 CFR 50.59, as described in Section 3.10 of NEI 96-07, Revision 1 and endorsed in Regulatory Guide 1.187, “Guidance for Implementation Of 10 CFR 50.59, ‘Changes, Tests, And Experiments’”.

#### 4.3.3.2 Regulatory Evaluation

The regulatory evaluation described in Section 4.3.1.2 of this report also applies to this subsection.

#### 4.3.3.3 Technical Evaluation

NEDC-34039P discusses applicability of TGBLA06/PANAC11 to the BWRX-300 design. The document presents TGBLA06/PANAC11 prediction-to-measurement comparisons for operating BWR plants. These include comparisons of core power distribution predictions with transversing incore probe (TIP) measurements. Plant-specific results are presented with respect to core-average exposure, core-average enrichment, power-to-flow ratio, nominal core power density, core-average exit void fraction, and maximum core exit void fraction for the operating plants and the BWRX-300. Compared to existing data, the BWRX-300 design is predicted to have a higher power-to-flow ratio and core exit void fraction, and a lower power density than plants for which measured data was provided. Additionally, based on information provided in PSAR Table 4.4-1, the CRN-1 core-average void fraction is predicted to be higher than void fractions in the qualification dataset.

Use of TGBLA06 and PANAC11 for void fractions comparable to those predicted for CRN-1 was discussed in the NRC staff safety evaluation of Supplement 6 to NEDC-33173P-A (ADAMS ML19228A268). This supplement requested modification of penalties previously imposed to address TGBLA06 and PANAC11 uncertainties, and the staff evaluation considered extrapolation to higher void fractions because operating data could not be obtained at MELLLA+ state points which would maximize void fraction. In the safety evaluation of this LTR supplement, the staff concluded that operation at these void fractions is not expected to lead to conditions that challenge GVH's neutronic methods or introduce increasing uncertainties based on trends in qualification data provided with the LTR supplement. However, because of the nature of the assessment, staff revised the NEDC-33173P-A limitations and conditions to require that licensees collect and analyze additional TIP data if operating at power-to-flow ratios exceeding those in the qualification data. The revised condition also requires licensees to institute appropriate measures to protect limits and submit the results to the NRC if any adverse trends or inconsistencies are observed.

In lieu of qualification data directly supporting use of TGBLA06 and PANAC11 for CRN-1 conditions, the same measurement, analysis, and reporting requirements are needed to confirm that these nuclear analysis codes achieve their expected performance, and these requirements should be described in the FSAR provided with the OLA. In the response to audit item A-4.3-1 (ML26091A346), TVA stated that a discussion concerning these reporting requirements (documented in limitation and condition 5 on NEDC-33173P-A) will be provided with the FSAR. The need to provide this information at the OL licensing phase and other LTR limitations and conditions is provided later in this report section.

NEDC-34039P quantifies uncertainties that will be used when TGBLA06 and PANAC11 are applied to BWRX-300. Definitions of uncertainty components and techniques to quantify these uncertainties are from historical GVH methods. The applicant provided justifications for values of uncertainty based on prior NRC approvals. The staff reviewed these justifications and finds that they are cited appropriately and support the BWRX-300 uncertainty quantification, subject to the items discussed in this section. The applicant also estimates core flow uncertainty based on system parameters because of the natural recirculation design of BWRX-300. The uncertainty estimate is larger than for operating BWRs. Methods of accident and transient analysis approved for BWR/2 through BWR/6 designs include approaches that account for variability in core flow and control rod position due to operational choices. This is discussed in additional detail below.

The staff considered application to the equilibrium cycle presented in PSAR Section 4.3; impacts of code qualification on the initial core loading and downstream analyses, such as on support for corrections to void reactivity coefficients, will be subject to NRC staff review with the application for an OL, as discussed in Section 4.3.4.3 of this SER.

Several components of uncertainty are dependent on the performance of the core monitoring system. Section 3.2.3 of NEDC-34039 states that core monitoring uncertainties were taken directly from NEDC-33197P-A, Revision 3, and TVA confirmed in the response to audit item A-4.3-3 (ML26091A346) that NEDC-33197P-A, Revision 3 supports the gamma thermometer design in the CRN-1 application. However, approval of the core monitoring methods described in NEDC-33197P-A is subject to adherence to limitations and conditions described in Section 4 of the NEDC-33197P-A safety evaluation, and alternative approaches may change the core monitoring uncertainty. In response to audit item A-4.3-3 (ML26091A346), TVA indicated that they intend to present and justify alternate approaches in the FSAR provided with the OLA. Therefore, the FSAR provided with the OLA should include (or reference an approved topical report that provides) a description of the gamma thermometer system, including LPRM calibration and power shape monitoring. Any deviations from a previously approved topical report or novel applications of the gamma thermometer system, including the items designated for completion with the FSAR in the response to CPA audit item A-4.3-3 (ML26091A346), shall be described and justified with respect to design basis functions and impact on core monitoring uncertainties.

NEDC-34039P did not identify approaches for identifying reasonable axial power shapes and exposure distributions that may exist at any given exposure point due to selection of control and burn strategies. Previously approved approaches (e.g., NEDE-32906P-A Supplement 3-A, Revision 1, or NEDC-33173P-A, Revision 5) consider multiple control strategies (e.g., deep control rods with possible flow reduction early in cycle, or shallow control rod insertion) that are intended to ensure limiting power shapes are analyzed. In response to a regulatory audit question that requested that the applicant identify whether these approaches would be applied to the BWRX-300 plant, the applicant cited Table 3-5 of NEDC-30434, which states that the analysis process is to cover the allowable range as appropriate. The applicant also indicated they may apply historical techniques to ensure the licensing analysis is bounding and results in an analysis envelope that the utility and GNF track through operation of the core. The staff considers this sufficient to support issuance of the CP, noting that the operating history assumed by the transient and accident methodologies is a potential subject of review with the OLA.

Appendix C of NEDC-34039P provides a methodology for evaluating $MCPR_{99.9\%}$. The staff's evaluation of MCPR operating limits is in Section 4.4 of this report. Appendix C of NEDC-34039P notes deviations from historical approaches, stating that because core flow uncertainty is determined based on uncertainties in other system characteristics, it would be inappropriate to sample underlying uncertainties used to define the core flow uncertainty. Staff understand that the applicant's proposed method may be sufficient, but the adequacy of this method must be demonstrated at the OL phase. This demonstration is expected to be associated with the additional information needed to address MCPR uncertainties provided in Section 4.4 of this report.

Because TGBLA06/PANAC11 have existing NRC approvals and a history of use in the domestic BWR fleet, NRC staff did not perform a de novo review of these methodologies. However, to confirm that these methodologies are being used in a manner consistent with previous NRC approvals, NRC staff requested that the applicant disposition limitations and conditions from relevant NRC-approved topical reports. The applicant provided a disposition of these limitations and conditions in response to audit item A-4.3-1 (ML26091A346). Staff reviewed this response and found the disposition of these limitations and conditions provide an acceptable basis for use of TGBLA06 and PANAC11 for analysis supporting issuance of a CP. The response notes that several limitations and conditions will be addressed with the FSAR that will be provided with the OLA. To ensure that the OLA is submitted with sufficient information for NRC staff to reach the requisite safety findings, the FSAR submitted with the OLA should include the following information:

The FSAR provided with the OLA shall address items designated for completion with the FSAR in response to CPA audit item A-4.3-1 (ML26091A346).

#### *4.3.3.4 Conclusion*

The NRC staff finds that TGBLA06/PANAC11, applied as described in NEDC-34039P and the NRC-approved topical reports referenced therein, is sufficiently applicable to BWRX-300 to produce analyses supporting authorization of construction. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

### 4.3.4 Core Nuclear Design Evaluation

#### *4.3.4.1 Introduction*

PSAR Section 4.3.4 summarizes results of analysis supporting the BWRX-300 reactor core nuclear design.

#### *4.3.4.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.3.1.2 of this report also applies to this subsection.

#### *4.3.4.3 Technical Evaluation*

PSAR Section 4.3.4.2 discusses reactivity feedback mechanisms and evaluation of specific reactivity coefficients. Demonstration that certain values of reactivity coefficients are bounded in certain situations can support compliance with GDC 11. NEDE-24011-P-A contains new fuel licensing acceptance criteria that restrict values of reactivity coefficients and reactivity feedback. NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” includes an evaluation of these requirements for GNF2 fuel applied to BWR/2 through BWR/6 designs licensed for operation in the United States. In contrast to domestic BWR/2 through BWR/6 plants, the CRN-1 core uses an N-lattice configuration, as specified in PSAR Section 4.3.6, which affects the fuel-to-moderator ratio and reactivity feedback. Accordingly, PSAR Section 3.1.2.2 specifies that separate confirmation of new fuel licensing acceptance criteria was performed for GNF2 fuel as applied to BWRX-300. NRC staff considers extension of the GNF2 evaluation against GESTAR II new fuel licensing acceptance criteria to the BWRX-300 N-lattice configuration sufficient to support issuance of the CP of CRN-1. These evaluations may be requested during NRC staff review of the FSAR provided with the OLA.

PSAR Figures 4.3-4 through 4.3-9 show results from the evaluation of the equilibrium cycle. Results include assembly-wise and axial power distributions at three cycle exposures.

PSAR Figure 4.3-7 shows the maximum calculated LHGR compared to a steady-state limit. PSAR Figure 4.3-9 shows maximum fraction of linear power density (MFLPD), which is the maximum fraction of linear heat generation rate with respect to the exposure-dependent thermal-mechanical limit. PSAR Figure 4.3-9 illustrates margin to thermal-mechanical limits during normal operation.

Figure 4.3-8 provides an evaluation of shutdown margin for the equilibrium cycle core. The results show that shutdown margin is evaluated for a range of cycle exposures, and that the equilibrium cycle meets the shutdown margin requirement specified in PSAR Section 4.3.4.4.

The staff finds these descriptions and evaluations of the equilibrium cycle design to be sufficient to support issuance of the CP of the CRN-1 facility. Generally, an equilibrium cycle is developed with the assumption that every cycle is operated identically and designed with the same shuffle pattern and feed fuel. Equilibrium core designs do not directly support demonstrations that actual core designs can be operated safely. Instead, they serve as reference core designs that are used to either demonstrate how cycle-specific analysis will be performed or to perform cycle-independent analyses of record that can be evaluated for applicability to future core designs using an approved methodology. Demonstration of the initial core will be needed for staff to evaluate the safety of the final CRN-1 design during review of the OLA. Therefore, the FSAR submitted with the OLA should address the following information:

The FSAR provided with the OLA should include a description of the CRN-1 initial cycle design (e.g., core loading pattern, operating control rod positions, and fuel design) and evaluations of initial core performance. The FSAR should include initial core design evaluations needed to determine margins of safety during normal operation and anticipated operational occurrences, including but not limited to margins to linear heat generation rate limits, margins to minimum critical power ratio limits, and shutdown margin. FSAR analyses that are dependent on core design should analyze the CRN-1 initial core.

The ability of reactivity control system to reliably control reactivity changes such that SAFDLs are not exceeded during type 1 AOOs (see Section 15.1.3.1 for explanation of type 1 AOOs and type 2 AOOs), and that core coolability is maintained during postulated accidents, depends on safety analyses evaluated in Chapter 15 of this report.

PSAR Section 4.3.4.9 includes discussion of load following operation. The impact on fuel thermal-mechanical analysis is discussed in PSAR Section 4.2. It notes that control rod adjustments are applied to maintain core power levels during xenon transients following core power maneuvers.

The staff did not evaluate non-baseload modes of operation during the review of the CPA, but considered information that may be needed to reach safety findings on the OLA. Although the PSAR does not specify an intended frequency of power changes, daily load-following power changes are assumed in the fatigue analysis for pressure-retaining components and fuel, as discussed in PSAR Section 3.9.1.1 and Section 4.2.3 of this SER. Staff notes that frequent power changes may undermine the assumption of equilibrium xenon conditions at initiation of design basis events, which may affect assumed initial control rod positions. Additionally, staff expects that deviation from the control rod positions and thermal-hydraulic conditions assumed in the core depletion analysis for a significant portion of the cycle would alter the core isotopic distribution and axial power shape. As part of a basis for non-baseload operation, the OLA should reference an application methodology that evaluates the effects of non-baseload operation on safety analysis initial conditions. While the equilibrium cycle provided in PSAR Section 4.3 is sufficient for the purposes of the CPA, future reference cycles may need to represent the intended range of non-baseload operation, accounting for fuel burnup and residence time limits. Therefore, the FSAR submitted with the OLA should address the following:

Non-baseload modes of operation were not approved in the CPA. If non-baseload operation is requested in the OLA, the FSAR must specify the desired mode of non-baseload operation, including the range of operating power levels and maximum frequency of power changes, and identify structures, systems, and components (SSCs) used to effect power changes. Safety analysis and fuel thermal-mechanical analysis should reflect the specified range of operation requested.

#### *4.3.4.4 Conclusion*

PSAR Section 4.3.4 includes information required by 10 CFR 50.34(a)(4). The evaluation provided shows that the equilibrium core design is consistent with criteria for nominal power distribution and shutdown margin, and margin to these limits is quantified. Staff will review the evaluation of the initial core design during review of the operating license application. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

### 4.3.5 Similarities to Previous Reactor Designs

#### *4.3.5.1 Introduction*

PSAR Section 4.3.5 relates certain aspects of the CRN-1 nuclear design to previous BWR designs.

#### *4.3.5.2 Regulatory Evaluation*

See Section 4.3.5.3.

#### *4.3.5.3 Technical Evaluation*

Information in PSAR Section 4.3.5 specifies certain features of the BWRX-300 and notes similarity to other BWR designs. The staff considered this information during its review of the CRN-1 CPA, but this information does not, in and of itself, provide the technical basis for a regulatory finding.

#### *4.3.5.4 Conclusion*

The staff considered information in this section during review of the CRN-1 CPA, but makes no specific regulatory finding relative to comparison of the BWRX-300 design to previous BWR designs.

### 4.3.6 Safety Evaluation

#### *4.3.6.1 Introduction*

In PSAR Section 4.3.6, the applicant lists regulatory requirements relevant to the nuclear design, and either summarizes or references information that supports compliance with these requirements, including how relevant GDCs and PDCs are met.

#### *4.3.6.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.3.1.2 of this report also applies to this subsection.

#### *4.3.6.3 Technical Evaluation*

Information in this section provides a summary of how GDCs and PDCs relevant to the nuclear design are met, including cross references to other PSAR sections, as applicable.

#### *4.3.6.4 Conclusion*

PSAR Section 4.3.6 helps to clarify the relationship between nuclear design bases and regulatory requirements pursuant to 10 CFR 50.34(a)(3)(ii).

### 4.3.7 Conclusion

As described above, PSAR Section 4.3 provides nuclear design information and example performance results for the purpose of authorizing construction. PSAR Section 4.3 identifies design basis functions and controlling parameters for the core design that are the result of nuclear analysis. It identifies high-level features of the CRN-1 core, describes an equilibrium cycle and provides equilibrium core performance with respect to design limits during steady full-power operation. The results show that the nominal core power distribution for the equilibrium cycle is within preliminary operating limits established to ensure cladding integrity. It describes the steady-state nuclear analysis codes that support development of controlling parameters. NRC staff considers this description of the nuclear design sufficient to support issuance of the CP pursuant to the regulations of 10 CFR 50.34(a) and 10 CFR 50.35(a), as applicable.

As described in the staff's safety evaluation above, the FSAR submitted with the OLA should address the items noted previously in this section.

## 4.4 Thermal and Hydraulic Design

### 4.4.1 Introduction

PSAR Section 4.4 describes the thermal and hydraulic design of the BWRX-300 reactor core. The section outlines the design bases and analytical methods used to establish core thermal-hydraulic safety limits and margin to core thermal limits. The analyses include the evaluation of flow distribution, pressure drop, and fuel to coolant heat transfer, and cite the codes and methodologies used to perform the analyses. The section establishes a core thermal-hydraulic design that will be used for AOO and DBA safety analysis in Chapter 15 of the PSAR.

### 4.4.2 Regulatory Evaluation

This section summarizes the regulatory requirements and guidance that provide the acceptance criteria for NRC staff review of the BWRX-300 thermal and hydraulic design basis.

**Applicable GDC from 10 CFR Part 50, Appendix A:**

**GDC 10 (Reactor Design):**

Requires that the reactor core and associated coolant, control, and protection systems be designed with sufficient margin to assure that SAFDLs are not exceeded during normal operation, including AOOs. The thermal and hydraulic design basis ensures that parameters like MCPR and LHGR are established to meet this requirement.

**GDC 12 (Suppression of Reactor Power Oscillations):**
Requires the core design to prevent power oscillations that could result in exceeding SAFDLs, or to reliably detect and suppress such oscillations. The thermal-hydraulic design basis supports stability and margin to oscillatory behavior. The GDC 12 evaluation is performed separately in the review of PSAR Appendix 4A (which is in SER Appendix 4A).

**Applicable Federal Regulations:**

**10 CFR 50.34 (Contents of Applications; Technical Information):**
Requires a detailed description of the thermal and hydraulic design basis, analytical methods, and compliance with relevant GDC in the PSAR.

**10 CFR 50.35 (Issuance of Construction Permits):**
Outlines the conditions under which a CP may be issued by the Commission, based on the review of a preliminary SAR describing a design which is not yet complete.

**Applicable Regulatory Guidance:**

**Regulatory Guide (RG) 1.70 (*Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*):**
Provides expectations for the content and level of detail for thermal and hydraulic design basis sections in the PSAR.

**NUREG-0800, Standard Review Plan (SRP):**

- **SRP Section 4.4, "Thermal and Hydraulic Design," Revision 2:** Provides acceptance criteria for establishing and evaluating thermal-hydraulic safety limits, analytical methods, and margin to fuel damage. PSAR Table 1.9-4 indicates conformance with this guidance.

**DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications"**

- This interim staff guidance (ISG) discusses some of the regulatory requirements for a CP, applicable review guidance in the SRP, and special topics related to CPAs. The appendix to this ISG supplements the SRP by clarifying the review of certain information in a CPA.

These requirements and guidance ensure that the thermal and hydraulic design basis described in the PSAR provides sufficient technical detail and regulatory foundation to demonstrate preliminary conformance to applicable safety standards for core cooling, power margin, and fuel integrity under all design basis and operational conditions. Information in PSAR Section 4.4 is used to support preliminary findings regarding 10 CFR Part 50 Appendix A GDC 15, 34, and 35, reviewed under other sections of this report.

### 4.4.3 Technical Evaluation

#### *4.4.3.1 Thermal and Hydraulic Design Basis*

PSAR Section 4.4.1 specifies two design parameters, CPR and LHGR, that are important in determining the safety limit of the nuclear and thermal-hydraulic design of the nuclear core region.

PSAR Section 4.4.1.3 describes the flow regime map in the applicant's safety analysis code TRACG, the use of which has been previously approved for application to safety analysis for operating U.S. BWRs. Examples of such prior approvals are: “GESTAR II” (NEDE-24011-P-A-31 and NEDE-24011-P-A-31-US) and “TRACG Application for ESBWR” (NEDC-33083P-A). Note that an upcoming update to GESTAR II will add BWRX-300 (as stated in “010N6956rA Engineering Report for Audit Question A-15.5-12.pdf,” which was audited [ML25191A113]). The staff reviewed the contents of GESTAR II Revision 31 (U.S. supplement) and compared and contrasted the proposed CRN-1 BWRX-300 preliminary design against previously approved BWR designs and find that it is reasonable to extend GESTAR II to cover BWRX-300, based on the content of GESTAR II, Revision 31. The flow regime map and correlations are adequate to calculate the core void distributions in TRACG calculations, but additional justification will be needed at the OL licensing phase for the calculation of void distributions in the large hydraulic diameter chimney region. The staff’s review of NEDE-34043P “BWRX-300 TRACG Application,” Revision 1, (ML25141A242, public) is described in Section 15.11 of this report.

The PSAR section describes hydraulics of the core as simulated in TRACG, where the input to the code contains information gleaned from full scale tests that measure the hydraulic characteristics of GNF2 fuel assemblies. The staff audited NEDC-34040P “BWRX-300 GNF2 Fuel Assembly Pressure Drop Characteristics,” which is referenced in PSAR Section 4.4, to confirm that the thermal-hydraulic models of the BWRX-300 are informed by quality relevant experimental data.

The PSAR section introduces core coolant flow distribution, stating that TRACG-computed core flows have a strong history of experimental validation by comparison to measurements made in the operating fleet of forced recirculation U.S. BWRs. Staff notes that while that legacy of validation is valuable, the BWRX-300 is a natural recirculation design. Section 4.4.3.5 of this report describes two audit interactions the staff conducted with the applicant and their vendor to seek extended validation for the natural recirculation flow drive of BWX, which represent different thermal-hydraulic conditions than those BWRs for which TRACG has demonstrated and quantified accuracy and reliability.

Fuel heat transfer physics is well modeled using TRACG. NEDE-32176P Revision 4 “TRACG Model Description” is effectively the TRACG theory manual; while not an approved LTR, NEDE-32176P documents the standard BWR thermal-hydraulic methods used in TRACG for fuel-to-coolant heat transfer.

The preliminary thermal and hydraulic design bases are established to be the standard for GVH BWR state-of-the-art design. The preliminary fuel selection, GNF2, is compatible with the thermal-hydraulics and neutronics code TRACG. The fuel and codes are what are used for operating BWRs, thoughBWRX-300 differs from operating designs (e.g., BWR/2-6) due to having a natural recirculation design. However, the different mode of circulation is not a problem if the mass flows of the BWRX-300 are in the applicable range. The cited design bases are all selected from the previously approved GESTAR II LTR (Revision 31), which generally applies to operating U.S. BWRs.

*4.4.3.2 Thermal and Hydraulic Analytical Methods*

PSAR Section 4.4.2 gives an overview of the analysis methods for hydraulics and heat transfer. While an overview of the methods of analysis for hydraulics and heat transfer is given in the PSAR, any detailed description of these methods is contained in documents referenced by the PSAR. The required methods include analytical tools to evaluate the fuel bundle critical power performance, the fuel cladding integrity safety limit, the 99.9% limit for MCPR, the Operating Limit Minimum Critical Power Ratio (OLMCPR), the LHGR operating limits, the void fraction distribution, the core pressure drop and hydraulic loads, the core coolant flow distribution, and the fuel thermal-mechanical performance. The analytical methods specified in the PSAR section 4.4 and its references generally have been applied and approved to previous reactor designs including the natural circulation ESBWR and they are applicable to the preliminary operating conditions in the BWRX-300.

The description of how those methods have been applied to the BWRX-300 is provided in "BWRX-300 TRACG Application" (NEDC-34043P, Revision 1). TRACG for BWRX-300 is a description of how TRACG, combined with other codes, is applied and why it is qualified to perform BWRX-300 calculations for LOCA and non-LOCA events. NEDC-34043P provides an overview of the historical development and code qualification of TRACG for operating reactors, SBWR, and ESBWR including references to the supporting documents. The report also describes how TRACG with links to other codes will be used for the BWRX-300 Chapter 15 analysis for AOOs, DBAs, and design extension condition events that do not involve core damage (events that are not severe accidents). This technical report has not previously been reviewed and approved by the staff so a more detailed review of TRACG for BWRX-300 (NEDC-34043P, Revision 1) is provided in Chapter 15 of this report (i.e., Section 15.11).

The core is to be protected during steady state operation, including AOOs, by maintaining an adequate margin to the safety limit minimum critical power ratio in transients by staying above the OLMCPR during normal operation. The critical power is the bundle power at which the limiting rod will reach the onset of a boiling transition away from nucleate boiling to post-critical-heat-flux (CHF) film boiling heat transfer. Preventing boiling transition will ensure that the SAFDLs are not exceeded during normal operation, including AOOs, because of a loss of adequate heat transfer. The BWRX-300 will use standard GNF2 fuel. The standard methodology for evaluating the fuel and its margin to the critical power limit is used for the BWRX-300 analysis in the PSAR.

For the OLMCPR, the applicant specifies the use of GEXL17 correlation, as described in PSAR reference NEDC-33292P, Revision 3, "GEXL17 Correlation for GNF2 Fuel." GEXL17 for GNF2 fuel is a part of the LTR "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," NEDC-33270P, Revision 11 (ML20244A106), is also reference in the PSAR. The GEXL17 correlation is appropriate for this design as it is already in use in the operating fleet of BWRs and boiling phenomena in the proposed BWRX-300 is very similar to the same in the operating fleet of BWRs. GEXL17 is a critical quality correlation capturing the ratio of the power at which a boiling transition occurs (the critical power) to the actual operating power. CPR primarily addresses "dryout," which is the disappearance of the liquid film on the fuel rod surface at high quality but the calculations can also predict CHF in regions of the bundle with lower void fractions. Dryout is a transition from annular flow (continuous liquid film around the rod) to dispersed flow (no continuous film, vapor and droplets only). Heat transfer in dispersed flow is significantly reduced, causing the cladding temperature to rise, but generally more gradually compared to departure from nucleate boiling in pressurized water reactors.

PSAR Section 4.4.2 cites an approved methodology for calculating fuel cladding integrity safety limit minimum critical power ratio (FCISLMCPR (Fuel Cladding Integrity Safety Limit Minimum Critical Power Ratio), an $MCPR_{95/95}$) (GESTAR II), which is justified for application to BWRX-300. During the regulatory audit, the staff was informed that an upcoming update to GESTAR II will incorporate BWRX-300 (ML26091A348 nonpublic; ML26091A346 public), "010N6956rA Engineering Report for Audit Question A-15.5-12.pdf" was read in audit).

The PSAR specifies the methods from the TRACG theory manual and the BWRX-300 TRACG application document that are used to determine OLMCPR upon consideration of analyses of AOOs in Chapter 15. The staff concludes these methods are consistent with GESTAR II and appropriate for application to BWRX-300.

PSAR Section 4.4.2 cross references PSAR Section 4.3 for the LHGR operating limit method specification and importantly specifies “margin to design limits for circumferential cladding strain and centerline fuel temperature is evaluated for AOOs in accordance with NEDC-34043P.”

PSAR Section 4.2.2 specifies PRIME as one methodology for calculating fuel heat transfer, but staff notes that PRIME is more generally used to predict mechanical fuel changes due to fuel heat transfer, which is discussed more appropriately in PSAR Section 4.2. Nevertheless, SE Section 4.4.3.1 discusses fuel-to-coolant heat transfer, which is being modeled using standard methods that have previously been approved by the staff. PRIME is a code with an established history and prior approval; for example, NEDC-33840P-A Revision 1, “The Prime Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance”) describes the basis for previous staff approval and information on capabilities of the code and limitations and conditions on its use. NEDC-34043P Revision 1 (which is referenced by the PSAR) allows for the following with respect to the relationship between PRIME and TRACG, “the TRACG time-dependent channel power history is used as input to PRIME…to calculate margin to fuel melt and cladding strain for AOO conditions. Typically, this is performed for the most limiting channel(s).” This is a conventional use of the code pair TRACG-PRIME and is consistent with staff expectations for an implementation of PRIME-TRACG coupling. Therefore, the staff determined that the coupling of the code pair TRACG-PRIME is appropriate for application to the BWRX-300 proposed design, which uses GNF2 fuel at low duty, which is to say the operating LHGR is far from the generic limit of the fuel design; specifically, the limiting LHGR is approximately 2.2 times greater than the LHGR for . Refer to PSAR Section 4.2.3.8 (and SE section 4.2.3) for cladding fatigue analysis considerations and PSAR Section 7.3.3.2 (and SE section 7.1.4) for a description on how the automatic thermal limits monitor (ATLM) provides protection against violating LHGR design limits.

#### *4.4.3.3 Thermal and Hydraulic Design Description*

PSAR Section 4.4.3 provides a high-level overview of the BWRX-300 thermal-hydraulic design. Section 4.4.3.1 mentions the RCS is natural circulation and that the flow rate is proportional to the core thermal power. The reason for the relationship between core thermal power and coolant flow rate is because coolant flow is driven in part by a net vapor generation rate, which is driven by core thermal power.

As a result of applicant interactions during the review (audit question response A-15.5-4FF, ML25191A113), the staff has been informed to expect further development of PSAR Figure 4.4-1 “BWRX-300 Power/Flow Map,” for the OLA. At present, it is the applicant position to not use variable feedwater heating/inlet subcooling as a means of reactivity control to adjust core thermal power or flow. The staff also expects that at the FSAR/OLA stage refinement of PSAR Table 4.4-2, “Expected Beginning of Cycle Total Core Flow and Nominal Feedwater Temperature versus Power” will have occurred. This table shows the variation in core flow percent and feedwater temperature as a function of core thermal power percent. Staff anticipates that measurement uncertainty, modeling uncertainty, steam system component design uncertainty and control system uncertainties will combine to form a narrow band around either the single natural circulation line on PSAR Figure 4.4-1 or the data provided in any adjustments to PSAR Table 4.4-2. Staff also expect there will be a variable core thermal power vs. RPV pressure band, similar in variation to the core thermal power vs feedwater temperature band. Stability, AOOs (vs. SAFDLS) and DBAs will need to be evaluated along the borders of the narrow operating bands for core flow percent and feedwater temperature and RPV pressure vs core thermal power (or the inverse) in order to obtain an OL for the facility; the staff will evaluate this information of the OL licensing phase.

PSAR Section 4.4.3.6 broaches the topic of load following, stating that because core power and core flow are loosely coupled—with core flow varying from 69.2 percent to 102.4 percent as core thermal power varies from 15 percent to 103 percent—compliance to thermal limits is expected to be maintained. Load following has not been reviewed or approved for the CP. Load following is addressed in Section 4.3.4.3.

*4.4.3.4 Thermal and Hydraulic Evaluation*

PSAR Section 4.4.4 provides information about important limits that constrain the thermal-hydraulics steady state design and how they are evaluated. The quantities were established for the operating fleet of BWRs, and the design and evaluation methods have been extended from the operating BWRs to the protype BWRX-300 design.

As expected from the use of GESTAR II, PSAR Section 4.4.4.1.1 specifies what it calls a "fuel cladding integrity safety limit" which is implemented as FCISLMCPR. 1.07 is a value of precedent for SLMCPR (Safety Limit Minimum Critical Power Ratio) using axial-varying R-Factor GEXL correlation; see GESTAR II NEDE–24011–P–A–31–US and NEDC-33270P Revision 11, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)" for details on the value of 1.07. The SLMCPR, to which the PSAR Chapter 15 computed limiting AOO ΔCPR is added to obtain the OLMCPR. According to PSAR SectionChapter 15 Table 15.7-1 "Results Summary of AOO Events," the preliminary limiting AOO ΔCPR occurs during "Closure of One Main Steam Reactor Isolation Valve AOO" which is a pressure increase event. However, the Chapter 15 SAFDL-limiting analysis was performed at nominal conditions, making the result sufficient only as a preliminary safety analysis.

PSAR Section 4.4.4.1.3 prescribes that OLMCPR will be calculated for each core and will be recorded in the COLR(s) (Core Operating Limit Report[s]). Also of importance, Section 4.4.4.2 declares that reference equilibrium cycle LHGR (in the form of MFLPD) recorded in NEDC--34044P "BWRX-300 GNF2 Equilibrium 12 Month Cycle Nuclear Design Report" Revision 0 (PSAR reference 4.4-10; ML25191A113) complies with the steady-state LHGR operating limit for the fuel; NEDC-34044P was audited by the staff during the review.

The methods employed in the calculation of simulation void fraction distributions are given by citation to the TRACG theory manual (NEDE-32176P revision 4) and are standard, state-of-the-art for GVH BWR designs. Based on staff experience with such LWR calculations, the determined that the use of TRACG is generally consistent with previous applications. However, the staff notes that PSAR Section 4.4.4.3 states that "higher power-to-flow ratios with associated higher exit void fraction conditions that will be observed in BWRX-300 compared to the current fleet should not be expected to increase power distribution uncertainties." This is a reasonable expectation on the part of the applicant and the power shape uncertainty stemming from voiding is consistent with the flatter radial variation in axial channel/bundle flows that the staff viewed in audit of steady state TRACG BWRX-300 simulation. Specifically, the natural circulation nature of the design leads to a more homogeneous distribution of core physics parameters. This is a qualitatively different conclusion for the TRACG hydrodynamic equations and correlations that relate to the radial void distribution in the BWRX *core*, which consists of channels and bypasses, versus the void distribution *in the chimney* (i.e., see Section 4.4.2.1 of this report).

*4.4.3.5 Evaluation of the Validity of Thermal and Hydraulic Design Techniques*

The evaluation in PSAR Section 4.4.4.5 is based on analytical design methods that have been approved for use under the same operating conditions as are expected and predicted for the BWRX-300, which is to say the conditions of operating U.S. BWRs. They have been validated against separate effects data, component level data, and operating plant data.

During the review, the staff audited the TRACG basedeck design basis record (DBR). The TRACG basedeck establishes the steady state for BWRX-300 as part of the confirmation of the model once it is developed. Key files viewed by the staff include summary files and associated steady state/null transient plots which are still evolving with the reactor design (i.e., these were preliminary results). The files and the information they contain serve as evidence of "…preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations…" (as stated in 10 CFR 50.34(a)(4)). Specifically, one of the DBR files contains the converged TRACG steady state full power results capturing the channels power, flow, and CPR, allowing for the radial variation of these quantitates to be observed. Also contained in the record is the radial variation in chimney void faction and core bypass exit void fraction. The axial void profile from the hottest fuel channel was viewed as well as total core flow, core inlet subcooling, recirculation flow rate and carry under fraction. The convergence results of the simulation with respect to operational targets such as dome pressure, steamline pressure drop, steam dryer pressure drop, and reactor water level (i.e., "maintain target level," DL2-02) were viewed. The record file also records comparison checks between TRACG and PANACEA, which is a steady state core simulator and cross section processing code.

During the audit, the staff requested the applicant to produce information to aid the staff in understanding the radial variation in the nominal, critical power ratio (CPR) figure of merit (FoM) margin to the MCPR SAFDL predicted for the BWRX-300 design. To understand the response of the reactor system to a transient, it is necessary to understand what the normal operation state looks like. This was to be done by quantitative comparison to those radial variations in conventional BWRs for fuel channels and discussion of the impact on the MCPR limit as flow is reduced. Staff also requested justification or evidence to support TRACG's accuracy in calculation of radial variation in inlet flow distribution in natural recirculation flows, which could come from representative experiments or validated computation fluid dynamics (CFD) software simulation

In response to staff questions during the regulatory audit, the applicant provided the following information:

The channel flow rates were used to compile the same type of information provided as part of an ESBWR RAI (Table 4.4-23-1 in ML063060119). The data was compiled for the core $\Delta p$, bypass flow, in-channel average flow and a standard deviation of the channel flow rates. This data was compared to the RAI table and the performance for BWRX-300 is similar to ESBWR. The standard deviation for BWRX-300 channel flow is lower than for the other BWR designs compared. Staff notes the applicant reported the following values:

| BWR Design | Channel Flow Standard Deviation | Number of Fuel Channels |
|---|---|---|
| BWRX-300 | 7.7 | 240 |
| BWR/6 (low flow) | 14.5 | 748 |
| BWR/4 (rated flow) | 15.8 | 764 |
| BWR/4 (low flow) | 16.1 | 764 |
| ESBWR | 11.1 | 1132 |

The applicant explained that these results were reasonable due to BWRX-300 having a smaller core with less potential variation in lower plenum and upper plenum boundary conditions. The staff find this to be a reasonable explanation for the radial flow variation predicted for BWRX-300. The applicant's vendor also remarked that although the average channel flows are similar, the BWRX-300 core pressure drop is larger than in ESBWR due to the use of standard fuel length GNF2 fuel compared to the shorter ESBWR fuel.

Additionally, the staff reviewed in audit processed output from experimentally validated CFD code simulation which the applicant's contractor (GVH) uses to substantiate the ability and reliability of TRACG to simulate coolant flow in natural recirculation BWR designs by making direct, quantitative comparisons between computed CFD results and computed TRACG results (i.e., the code-to-code difference in a computed total pressure at one or two locations in a hypothetical BWRX-300 design). The code-to-code comparison with an experimental comparison for the CFD shows TRACG adequately simulates steady state physics based CFD results. Information from DBR-0071125 R0 and DBR-0069238 R0 show very small radial pressure drop on the average through the axial extent (i.e., the upper and lower plenums) of the core, which the applicant's vendor states is consistent with TRACG prediction. As a result of the audit interactions with the applicant and their vendor regarding experimental and/or CFD comparisons for TRACG of the natural recirculation hypothetical BWRX-300, the staff gained confidence in the ability of a code (TRACG), which has extensive history for forced recirculation designs, to adequately and reliably simulate the important, governing aspects of BWRX-300 for steady state and transient conditions.

#### *4.4.3.6 Core Monitoring*

PSAR Section 4.4.6 notes that the 3-D core thermal power distribution monitor is described in PSAR Section 7.3. It is part of the instrumentation and control system and is discussed in Section 7.1.4 of this report.

#### *4.4.3.7 Summary of Technical Evaluation*

The staff concludes that the thermal-hydraulic design of the reactor core is suitable for preliminary analysis against the requirements of GDC 10 and 12 of Appendix A to 10 CFR

Part 50. The definitions of, and methods to compute, SAFDLs of CPR, which covers concepts like OLMCPR vs. SLMCPR and 95/95 values vs. 99.9 percent values, and MFLPD (which is LHGR, in a way) were further expounded in PSAR Section 4.4 subsequent to their introduction in PSAR Section 4.3. The methods of calculating these two quantities and the margin of the state of the reactor to those two limits are not new and are suitable for application to the BWRX-300 CRN-1 CP proposal.

Regarding what SRP Section 4.4 Revision 2 considers to be the ten specific acceptance criteria necessary to meet the requirements of GDC 10 and GDC 12, the applicant presented their own conformance review in PSAR Table 1.9-4 "Conformance with NUREG-0800 (Chapter 4 Reactor)." According to this PSAR table, the only SRP 4.4 criteria that appear in PSAR Section 4.4 are 1, 2, 4, 5, and 8. Criterion 1 addresses uncertainty in the state of process parameters and instrumentation as it affects margin to SAFDLs, which is adequately handled by the phenomena identification and ranking table multipliers in TRACG (see Section 15.11 of this report for the review of NEDC-34043P "BWRX-300 TRACG Application," Revision 1). Criterion 2 addresses uncertainty in SAFDL quantities, which is adequately addressed by the applicant via citation of GESTAR II Revision 31. The staff notes that in auditing "010N6956rA Engineering Report for Audit Question A-15.5-12.pdf," the applicant's contractor states there is an upcoming revision for GESTAR II which will incorporate the BWRX-300 design explicitly into the fuel reload licensing report. The radial variation core flow distribution was scrutinized during the audit, as captured in Section 4.4.3.5 of this report.

SRP 4.4, Criterion 4 addresses the quality of calculating the phasic state of the boiling water coolant; specifically the SRP criterion states "[m]ethods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components should include classical fluid mechanics relationships and appropriate empirical correlations." The applicant uses the computer code TRACG to do this, and this code, the equations it contains, and empirical correlations was written specifically to model fluid flow (static and dynamic) and heat transfer in BWRs. This code has been approved for operating BWRs and the single- and two-phase fluid flows in BWRX are governed by the same physics processes that apply to the operating BWRs. Staff's review of the application of TRACG to BWRX is found in section 15.11 of this report. SRP 4.4, Criterion 5 addresses technical specifications specific to SAFDLs in AOOs; as technical specifications are yet to be developed, the staff can only say that the applicant has used standard BWR methods and has calculated preliminary operating limits such that SAFDLs are protected during normal operations and AOOs, with the caveats in the next paragraph. SRP 4.4, Criterion 8 is a further specified consideration of RCS pressure drop uncertainty which is adequately handled in the way TRACG is used (phenomena identification and ranking table multipliers) and has been rigorously assessed in PSAR reference 4.4-3: NEDC-34040P Revision 0, "BWRX-300 GNF2 Fuel Assembly Pressure Drop Characteristics," which the staff audited.

The results in PSAR Chapter 15 indicate that the preliminary BWRX-300 design is can reasonably comply with GDC 10. However, the PSAR Chapter 15 SAFDL-limiting analysis was performed at nominal conditions, with a systematic implementation of single failure assumptions and a range of electrical power availability conditions. These preliminary safety analyses are sufficient for the purposes of supporting the issuance of a CP, but these limitations will need to be addressed at the OL phase. Preliminary compliance GDC 12 will be considered in Appendix 4A of this report; SRP 4.4 Acceptance Criterion 3 is also addressed in the appendix to this section. SRP 4.4 Criterion 6 applies to 10 CFR Part 52 applications and is not applicable. SRP 4.4 Criterion 7 addresses loose parts monitoring systems which are not included in the BWRX-300 design. SRP 4.4, Criterion 9 addresses inadequate core cooling instrumentation which the applicant has deferred to FSAR. SRP 4.4 Criterion 10 addresses anticipated transient without scram (ATWS) analysis of which is deferred to FSAR, consistent with 10 CFR 50.62(d).

### 4.4.4 Conclusion

The staff has reviewed the applicable information in the PSAR Section 4.4 and other references and has determined that it meets the regulatory requirements identified in this SE section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a) and 10 CFR 50.35 as applicable. Staff notes that because the PSAR Chapter 15 SAFDL-limiting analyses were performed starting from nominal conditions with no biases or uncertainty considered, the OLMCPR computed via the methods described and referred to in PSAR Section 4.4 is preliminary, which is adequate for the BWRX-300 CPA. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

Additional technical information that will need to be addressed in the OL are listed below:

- Stability, AOOs, and DBAs need to be evaluated along the borders of the narrow operating bands for core flow percent and feedwater temperature and RPV pressure vs. core thermal power (or the inverse).
- The FSAR should include a listing of the nuclear codes, methods and analytical results (within either technical or topical reports) that are either incorporated by reference, or are described, outlined or summarized in sufficient detail to be considered described within FSAR for the purpose of applying 10 CFR 50.59, as described in Section 3.10 of NEI 96-07, Revision 1, as endorsed by RG 1.187.
- The applicant should provide full uncertainty/bias treatment for MCPR/LHGR, including OLMCPR and limiting ΔCPR with uncertainties, at FSAR using approved methodologies.
- The inadequate core cooling instrumentation must be specified in the FSAR at the OLA stage and should address the information requested in Generic Letters 84-23 and 92-04, as applicable.
- GESTAR II should be updated to include BWRX-300.

## 4.5 Reactor Materials

PSAR Section 4.5 describes the materials used in the CRD system and reactor internals for the BWRX-300. The section addresses material selection, specification, fabrication, and quality assurance practices for both the CRD system structural materials and the reactor internals materials. For the CRD system, 300 series austenitic stainless steels, nickel-chromium-iron alloys (such as Alloy X-750), XM-19, and 17-4 PH stainless steels are used, chosen for their corrosion resistance, mechanical properties, and proven performance in BWR environments. The RCPB components are fabricated in accordance with ASME BPVC Section III. The section also describes controls on fabrication, welding, heat treatment, and cleanliness to prevent stress corrosion cracking and ensure long-term material integrity.

For reactor internals, austenitic stainless steels are primarily used to minimize activation and corrosion, with nickel-base alloys or precipitation-hardened steels utilized in specific locations requiring higher strength or special properties. PSAR Section 4.5 also provides information on material compliance with GDC, ASME BPVC, and RGs, as well as quality assurance, cleaning, inspection, and controls for seismic, dynamic, and environmental loadings. The material selection and fabrication processes are presented as supporting reliable operation, resistance to degradation, and structural integrity throughout the plant's design life.

### 4.5.1 Control Rod Drive System Structural Materials

#### *4.5.1.1 Introduction*

The PSAR, Revision 1, as supplemented by letters dated October 1, 2025 (ML25275A436) and December 2, 2025 (ML25336A282), in regards to response to audit questions 4.5.1-1, 4.5.1-2 and 4.5.1-3), describes the materials used in the control rod drive mechanism (CRDM) for both the reactor coolant pressure boundary (RCPB) portion of the CRDM and non-pressure boundary CRDM components. The information provided in PSAR Section 4.5.1, and the supplements include material specifications for base material and weld metal (austenitic stainless steel and other materials), fabrication and processing of austenitic stainless-steel components, and cleanliness control.

#### *4.5.1.2 Regulatory Evaluation*

The staff based its review on the relevant requirements of the following regulations and provisions of the following guidance:

GDC 1, "Quality standards and records," and 10 CFR 50.55a(a)(1) require, in part, that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. These quality standards shall be identified and evaluated to determine their adequacy to ensure a quality product, in keeping with the required safety function.

GDC 14, "Reactor coolant pressure boundary," requires that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

GDC 26 requires, in part, that one of the radioactivity control systems shall use control rods (preferably including a positive means for inserting the rods) and shall be capable of reliably controlling reactivity changes so that SAFDLs are not exceeded under conditions of normal operation, including AOOs. In PSAR Section 3.1.3.7, the applicant identified an alternative PDC 26 to address reactivity control system redundancy.

RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," provides the degree of surface cleanliness.

RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," provides acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless-steel components.

RG 1.44, "Control of the Use of Sensitized Stainless Steel," provides guidance for avoiding sensitization in stainless steel through the use of reduced carbon content and process controls.

RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," provides guidance for preparation of PSAR Section 4.5.1 for a CPA.

RG 1.71, "Welder Qualification for Areas of Limited Accessibility" provides additional performance qualification and requalification guidance for welding in areas of limited access.

As described below, the staff reviewed the structural materials aspects of the CRD in accordance with the regulatory guidance and standards to meet the regulations.

*4.5.1.3 Technical Evaluation*

The staff reviewed and evaluated the information in PSAR Section 4.5.1, as supplemented in response to Audit Questions 4.5-1 and A-4.5-2 (ML25275A438) and Audit Question 4.5-3 (ML25336A283), to ensure that the materials specifications, fabrication, and process controls were consistent with the criteria of SRP Section 4.5.1 and RG 1.70 to meet the requirements of GDC 1 and 14, and PDC 26. The staff determined that the applicant provided sufficient preliminary information in the PSAR in accordance with the guidance of RG 1.70 to support issuance of a CP.

Materials Specifications

The staff reviewed PSAR Section 4.5.1 to determine the suitability of the materials for this application. The PSAR provides information on the specifications, types, grades, heat treatments, and properties used for the materials of the CRD components.

The CRD structural components that are part of the RCPB include the fine motion control rod drive (FMCRD) middle flange including the ball check valve, and the lower component housing, which encloses the lower part of the drive, and are fabricated from austenitic stainless-steel forgings (SA-965 or SA-182 F304/F304L/F316/F316L). The mounting bolts are SA-193, Grade B7. These materials comply with the requirements in the ASME Code, Sections II and III, and are acceptable for use in the BWRX-300 design.

The remaining structural components identified in Table 4.5-2 of the PSAR are not part of the RCPB. The PSAR Section 4.5.1 indicates that the properties of those components are equivalent to those given in Parts A, B, and D of Section II of the ASME Code or those included in RG 1.84, “Design, Fabrication and Materials Code Case Acceptability, ASME Section III,” and are therefore acceptable for use in CRD components.

Austenitic Stainless-Steel Components

The applicant indicated that all stainless-steel materials are used in the solution heat-treated condition (as provided in ML25275A435). For all welded components exposed to service temperatures exceeding 93°C, the carbon content in the austenitic stainless-steel components is limited, not to exceed 0.020 percent. Limiting the carbon content in welded components experiencing service temperatures exceeding 93°C to 0.020 percent or less is consistent with NUREG-0313, Revision 2, “Technical Report on Materials Selection Processing Guidelines for BWR Coolant Pressure Boundary Piping,” and SRP Section 4.5.1. In addition, all filler metal for stainless steel will be low carbon with a maximum carbon content of 0.035 percent, which is consistent with GL 88-01. The staff finds this acceptable because the applicant will follow the guidance in GL 88-01 and NUREG-0313 to reduce the susceptibility of components to stress corrosion cracking (SCC).

The applicant indicated that cold-worked 300 series austenitic stainless steels are not used except for minor forming and straightening. However, if minor forming and straightening are performed, the process will be controlled by limiting the material hardness, bend radius, or the amount of strain induced by the process, and the maximum yield strength used would be 90 ksi, in conformance with RG 1.44. These controls are also applied to abrasive grinding. The staff considers the applicant’s cold-work controls for all stainless-steel components in the reactor system adequate to reduce the susceptibility of stainless-steel materials to SCC resulting from cold working including grinding.

The applicant indicated that the BWRX-300 design complies with RG 1.44, which provides the acceptance criteria for testing, alloy compositions, welding, heat treatment, cleaning, and protecting austenitic stainless steels to avoid severe sensitization. RG 1.44 specifies the test used to detect susceptibility to intergranular attack is in accordance with American Society for Testing and Materials (ASTM) A 262, “Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels”. The staff finds this acceptable because the applicant will conform to the guidelines in RG 1.44 as listed in SRP Section 4.5.1.

RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” provides the acceptance criteria for delta ferrite in austenitic stainless welds. These acceptance criteria address the recommended range of delta ferrite in stainless steel weld metal to avoid microfissuring in welds. The RG also includes a recommended procedure for ferrite measurement (Audit questions 4.5.1-3 and 4.5.1-1 requested whether RG 1.31 will be followed). The staff finds this acceptable because the applicant will follow the guidance in RG 1.31 to minimize the presence of microfissures in austenitic stainless-steel welds.

Other Materials

PSAR Table 4.5-2 states that to reduce risk of cracking, only the H1075, H1100, and H1150 heat treat conditions are allowed in 17-4 PH /Type 630 stainless steel for the ball nut. Based on industry experience using X-750 in CRD components and the guidance in SRP Section 4.5.1, the staff finds the use of Alloy X-750 for CRD components acceptable.

The use of Alloy X-750 for the coupling spud components is consistent with the Type 3 heat treatment of ASME SB-637, “Specification for Precipitation-Hardening Nickel Alloy Bars, Forgings, and Forging Stock for High-Temperature Service.” The high-temperature anneal treatment and the Rc40 maximum hardness specified in ASME SB-637 for the Type 3 heat treatment provides adequate stress-corrosion resistance in X-750 for BWR environments. Therefore, based on industry experience using X-750 in CRD components, the staff finds the applicant’s use of Alloy X-750 for CRD components acceptable.

Cleaning and Cleanliness Control

PSAR Section 4.5.1.4, “Cleaning and Cleanliness Control,” specifies that the cleaning and cleanliness controls for the CRDM during manufacture and assembly will meets the guidance in RG 1.28 and ASME NQA-1-2015. SRP Section 4.5.1 recommends that cleaning and cleanliness controls for CRDMs should be implemented in accordance with ASME NQA 1, which has strict process controls for cleaning and protection against contamination of materials during all stages of component manufacture and installation. For example, tools used in abrasive work on austenitic stainless steel, such as grinding, should not contain and should not have been contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or SCC. Therefore, the staff finds the applicant’s cleaning and cleanliness controls for CRDM components acceptable and consistent with SRP Section 4.5.1.

### *4.5.1.4 Conclusion*

The staff believes the applicant provided sufficient information on the selection of materials, fabrication processes, compatibility of materials, and cleaning and cleanliness controls to be acceptable because they satisfy regulatory requirements or positions described above (for RCPB materials), or because they have been demonstrated to be acceptable based on appropriate materials selections and acceptable operating experience (for non-RCPB materials).

Based on the above, the staff concludes that the design of the CRD structural materials is acceptable and meets the requirements of GDC 1 and 14, and PDC 26, as well as 10 CFR 50.55a. Therefore, the NRC staff finds that the information in PSAR Section 4.5.1 related to control rod drive system structural materials is sufficient and meets the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a) and 50.35, as applicable.

### 4.5.2 Reactor Internals Materials

#### *4.5.2.1 Introduction*

PSAR Section 4.5.3 discusses the materials selected for the reactor internals of the BWRX-300. The section identifies the principal use of austenitic stainless steels, and nickel-based and precipitation-hardened alloys where higher strength or enhanced corrosion resistance is required. The PSAR states that material specifications, fabrication, welding, and cleaning processes are aligned with ASME Code and NRC-endorsed guidance to ensure long-term integrity and minimize potential for degradation such as SCC.

#### *4.5.2.2 Regulatory Evaluation*

The NRC staff review of the reactor internals materials in Section 4.5.3 is based on the following regulatory requirements and guidance documents, which collectively form the acceptance criteria for this section:

**Applicable Regulations and GDC:**

**10 CFR 50.55a, “Codes and Standards”:**
Requires that reactor internals be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the safety importance of their function, using NRC-endorsed codes and standards.

**10 CFR Part 50, Appendix A, General Design Criterion 1, “Quality Standards and Records”:**
Requires that SSCs important to safety, including reactor internals, be designed, fabricated, erected, and tested to quality standards commensurate with the safety function to be performed, with adequate records maintained.

**GDC 4, “Environmental and Dynamic Effects Design Bases”:**
Pertains to protection against dynamic effects such as pipe breaks and environmental conditions, ensuring material selection and construction practices are adequate to maintain integrity.

**GDC 14, “Reactor Coolant Pressure Boundary”:**
Pertains to that the reactor coolant pressure boundary, including relevant reactor internals, be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

**Supporting Regulatory Requirements:**

**10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants":**
Establishes quality assurance program requirements for design, procurement, fabrication, inspection, and testing of safety-related SSCs.

**Regulatory Guidance, Codes, and Standards:**

**ASME *Boiler and Pressure Vessel Code* (BPVC), Section II, "Materials" and Section III, "Rules for Construction of Nuclear Facility Components":**
Provides the baseline for material specifications, fabrication, and welding of reactor internals.

**ASME BPVC, Section IX, "Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators":**
Governs the qualification of welding procedures and personnel involved in fabrication.

**ASME BPVC, Section XI, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants":**
Addresses inspection and testing requirements for in-service reactor vessel internals.

**Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal":**
Provides acceptance criteria and procedures for controlling ferrite content in stainless steel welds to minimize SCC and microfissuring.

**RG 1.44, "Control of the Use of Sensitized Stainless Steel":**
Specifies controls for sensitization, cleaning, and protection of austenitic stainless steel to reduce the likelihood of intergranular corrosion and SCC.

**RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III":**
Identifies acceptable ASME Code Cases for materials used in nuclear applications.

**ASTM A262, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels":**
Defines test methods for verifying resistance to intergranular attack as part of materials acceptance.

**SRP Section 4.5.2, "Reactor Internal and Core Support Structure Materials":**
Provides NRC staff acceptance criteria and review procedures for reactor internal materials, including the use of low-cobalt and SCC-resistant alloys, proper fabrication, and cleaning practices.

**RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants":**
Specifies content and format for PSAR sections addressing material selection and fabrication.

**Description and Relevance of Key Requirements:**

**GDC 1** establishes the fundamental expectation for high-quality standards and records for all safety-related materials and components.

**GDC 4** sets expectations that selected materials and fabrication methods maintain structural integrity under all credible environmental and dynamic conditions.

**GDC 14** extends high-integrity requirements to all pressure boundary and closely associated internal components.

**10 CFR 50.55a** and the ASME Code ensure that material selection, qualification, and fabrication processes are consistent with industry best practices and NRC-endorsed standards.

**RGs 1.31, 1.44, and 1.84** provide the technical bases for staff acceptance of applicant practices regarding ferrite control, prevention of sensitization, and code case applicability, respectively.

In summary, the staff uses these regulations, criteria, and guidance documents to determine whether the applicant's selection, specification, fabrication, and quality assurance practices for reactor internals materials are sufficient to ensure long-term structural integrity, minimize the potential for in-service degradation (including SCC), and support safe plant operation throughout the design life of the facility.

### *4.5.2.3 Technical Evaluation*

The staff divided its evaluation of the discussion on reactor vessel internals (RVI) and core support materials in the PSAR Section 4.5.3 into four topics, mapped to those described in SRP Section 4.5.2: (1) materials specifications, (2) controls on welding, (3) nondestructive examination, and (4) fabrication and processing of austenitic stainless-steel components.

#### *4.5.2.3.1 Materials Specifications*

In PSAR Section 4.5.3.1, the application specifies that core support materials will satisfy the requirements of ASME Code, Section III, Subarticle NG-2120, and the applicable requirements of ASME Code, Section II, Part D, Tables 2A, 2B, and Code Case N-60-6. The remaining portions of the RVIs are designed to conform to ASME Code, Section III, Article NG-1122. The staff finds this to be acceptable because it complies with the ASME Code and 10 CFR 50.55a.

#### *4.5.2.3.2 Controls on Welding*

The staff reviewed the controls on welding in PSAR Section 4.5.3.2, "Controls on Welding," specifically, the citations of ASME Code sections; RG 1.71 guidance; Section 5.2.3 information. The staff found the information presented acceptable because it complies with the SRP criteria for this topic.

#### *4.5.2.3.3 Nondestructive Examination*

The staff reviewed the nondestructive examination information in PSAR Section 4.5.3.3, "Nondestructive Examination," specifically, the citation of ASME Code sections. The staff found the information presented acceptable because it complies with the SRP criteria for this topic.

#### *4.5.2.3.4 Fabrication and Processing of Austenitic Stainless-Steel Components*

The staff reviewed PSAR Section 4.5.2.3.4, with emphasis on heat treatment, controls on sensitization, compatibility with reactor coolant, abrasive work, and minimization of contamination. The staff confirmed that the applicant noted appropriate controls on heat treatments. The staff confirmed that environmental conditions are controlled and that welding procedures are developed to minimize the probability of sensitization and microfissuring. This is achieved by following the guidance of RG 1.44 and RG 1.31, respectively. The staff confirmed the RVI and core support material compatibility with coolant through a review of the selection of materials for each component; consistent with RGs and ASME Code requirements; the topics detailed in PSAR Sections 4.5.3.4 and 5.2.3.4, "Fabrication and Processing of Austenitic

Stainless Steels." The staff reviewed the fabrication and cleaning controls imposed on stainless steel components and found them acceptable because they allow no contamination with ferritic or other incompatible materials and subsequent usage on austenitic materials. Because the fabrication, processing, and cleaning controls conform to the recommendations and requirements of the ASME Code and RG 1.31, the staff concludes that they are acceptable.

#### *4.5.2.4 Conclusion*

The staff concludes that the materials used for the reactor internals and core support structures are acceptable and meet the requirements of 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1. This conclusion is based upon the considerations discussed below.

The applicant has selected, and identified by specification, materials for the reactor internals and core support structures that satisfy the requirements of Sub-article NG-2120 of Section III, Division 1 and Tables 2A, 2B and 4 of Section II of the ASME Code. For materials not in accordance with ASME Code provisions, the applicant has selected materials of construction that are approved for use by NRC-accepted ASME Code Cases, as identified in RG 1.84, or that have otherwise been demonstrated acceptable for the application. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible.

The applicant has demonstrated that the design, fabrication, and testing of the materials used in the reactor internals and core support structures are of high-quality standards and are adequate to assure structural integrity. The controls imposed upon austenitic SSCs satisfy the positions of RG 1.44, "Control of the Use of Sensitized Stainless Steel," and the related criteria provided in SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

The controls imposed on the reactor coolant chemistry provide reasonable assurance that the reactor internals and core support structures will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of component structural integrity.

The material selection, fabrication practices, examination and testing procedures, and control practices provide reasonable assurance that the materials used for the reactor internals and core support structures will be in a metallurgical condition that will preclude in-service deterioration.

Conformance with relevant requirements of the ASME Code, or accepted Code Cases, and the recommendations of RGs 1.31 and 1.44 and the related criteria in SRP Section 5.2.3, constitutes an acceptable basis for meeting the relevant requirements of 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

## 4.6 Design of Reactivity Control Systems

PSAR Section 4.6 provides part of the design basis for, and description of, the CRD system.

### 4.6.1 Control Rod Drive System Design Bases

#### *4.6.1.1 Introduction*

This PSAR section summarizes the high-level design principles of the CRD system. It notes that the CRD system provides the primary means for reactivity control during normal, abnormal, and accident conditions, and that the design of the CRD system provides two motive forces for inserting control rods which rely on different principles, i.e., control rod insertion through injection of high-pressure water and motor-driven control rod insertion. It states that the design includes features designed to prevent unintended movement or malfunction of control rods.

#### *4.6.1.2 Regulatory Evaluation*

The staff reviewed PSAR Section 4.6 in accordance with the regulatory guidance for the review of CRD systems such as NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (hereafter referred to as the SRP), Section 4.6, Revision 2, issued March 2007. Staff's review of the control rod drive system (CRDS) was focused on CRD system functions and the relation to applicable regulatory criteria:

10 CFR 50, Appendix A GDC 4, "Environmental and dynamic effects design bases," as related to the environmental conditions caused by high- or moderate-energy pipe breaks during normal plant operation as well as postulated accidents.

GDC 23, "Protection system failure modes," as related to failure of this system placing the reactor in a safe state.

GDC 25, "Protection system requirements for reactivity control malfunctions," as related to the functional design of protection systems to ensure that SAFDLs are not exceeded for a single malfunction of the reactivity control system."

GDC 26, "Reactivity control system redundancy and capability," as related to the capability of reactivity control systems to control reactivity changes resulting from normal operations, including AOOs, such that SAFDLs are not exceeded. In PSAR Section 3.1.3.7, the applicant identified an alternative PDC 26 to address reactivity control system redundancy.

GDC 27, "Combined reactivity control systems capability," as related to the capability of reactivity control systems to reliably control reactivity changes under postulated accident conditions such that the capability to cool the core is maintained.

GDC 28, as related to limits on the potential amount and rate of reactivity increase to assure that acceptable postulated reactivity accidents do not result in loss of the capability to cool the core or damage to the reactor coolant pressure boundary greater than limited local yielding. In PSAR Section 3.1.3.9, the applicant identified an alternative PDC 28 to address reactivity limits.

GDC 29, "Protection against anticipated operational occurrences," as related to the reliability of reactivity control systems to accomplish their safety functions in the event of anticipated operational occurrences.

10 CFR 50.62(c)(3), as related to the diversity of the alternate rod injection system and redundancy of scram air header exhaust valves.

GDC 33, "Reactor Coolant Makeup," as related to the provision of a system to supply reactor coolant makeup such that SAFDLs are not exceeded due to leakage from, or small breaks in, the reactor coolant pressure boundary.

10 CFR 50.2 as it defines safety-related SSCs

#### *4.6.1.3 Technical Evaluation*

Section 4.6.1 presents high level design bases of the CRD system. Although this PSAR Section is titled "Control Rod Drive System Design Bases," NRC staff notes that design bases (as defined by 10 CFR 50.2) of the CRD system are located in this and other subsections of PSAR Section 4.6. For example, PSAR Section 4.6.2 lists design basis functions of the CRD system.

#### *4.6.1.4 Conclusion*

The NRC staff finds that the PSAR provides the design bases (as defined by 10 CFR 50.2) of the CRD system as part of the preliminary CRN-1 design, consistent with requirements of 10 CFR 50.34(a)(3)(ii). NRC staff review of the relationship between CRD system design bases and applicable principal design criteria is summarized throughout PSAR Section 4.6 and its subsections.

### 4.6.2 Control Rod Drive System Functions and System Classification

#### *4.6.2.1 Introduction*

PSAR Section 4.6.2 lists functions of the CRDS. Functions are assigned a safety category, either Safety Category 1, 2, 3, or N. This categorization is described throughout PSAR Section 3.2. It states that components required to perform Safety Category 1, 2, or 3 functions are assigned to Safety Class 1 (SC1), at least SC2, or at least SC3, respectively. Other components are assigned to Non-safety Class (SCN). The SSC classification of the CRD components are also listed in Table 3A-1. As described in Section 5.1 of NEDC-33934, Safety Category 1 categorization is equivalent to "safety-related" as defined in 10 CFR 50.2. Table 5-1 of NEDC-33934 indicates that Safety Category 2 and Safety Category 3 include "important-to-safety" components (as used in Appendix A to 10 CFR Part 50), "non-safety-related" components, and components that may meet some criteria for the regulatory treatment of non-safety systems.

In addition to these functions, PSAR Section 3.1.4.4 cites the SC3 high-pressure makeup function of the CRD system to assure that SAFDLs are not exceeded due to leakage or small breaks in the reactor coolant pressure boundary. The applicant credits this function of the CRD makeup system to meet GDC 33.

#### *4.6.2.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.6.1.2 also applies to this subsection.

10 CFR 50.2 defines safety-related SSCs.

#### *4.6.2.3 Technical Evaluation*

The staff reviewed the design basis functions of the CRD system listed in PSAR Section 4.6.2. NRC staff identified that other PSAR sections appear to rely on CRD functions that are not clearly identified as safety functions in PSAR Section 4.6.2. For example, the CRD hydraulic subsystem is credited in providing CRD makeup for small unisolable leak events, and is discussed in PSAR Section 3.1.4.4 as a means for meeting GDC 33. The NRC staff considers systems that satisfy the reactor coolant makeup function required by GDC 33 to be important to safety and thus subject to GDC 1, 2, and 4, as applicable for the function they are performing. This issue and the SSC classification is discussed and evaluated in Section 4.6.6 of this report.

#### *4.6.2.4 Conclusion*

Conclusions concerning the design basis functions of the CRD system are provided in Section 4.6.6 of this report.

### 4.6.3 Safety Evaluation

#### *4.6.3.1 Introduction*

In PSAR Section 4.6.4, the applicant lists regulatory requirements relevant to the CRD design, and references information that discusses compliance with these requirements.

#### *4.6.3.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.6.1.2 also applies to this subsection.

#### *4.6.3.3 Technical Evaluation*

Information in this section links technical content in other PSAR sections to some design criteria and regulatory requirements relevant to the CRD design. As this section does not cross-reference to PSAR Section 4.6, which includes CRD design bases, NRC staff understands that this section provides additional information to clarify how the design meets regulatory requirements.

#### *4.6.3.4 Conclusion*

This section provides information that supplements PSAR Section 4.6 information on the relationship between design bases and design criteria for the preliminary design pursuant to 10 CFR 50.34(a)(3)(ii). Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

### 4.6.4 Testing and Inspection Requirements

#### *4.6.4.1 Introduction*

PSAR Section 4.6.5 indicates that preoperational and startup testing is performed by the Initial Test Program (Chapter 14). See the safety evaluation for Chapter 14 for details. Detailed testing procedures are to be provided in the FSAR.

#### *4.6.4.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.6.1.2 also applies to this subsection.

#### *4.6.4.3 Technical Evaluation*

Technical information will be provided for NRC staff review with the OLA.

#### 4.6.4.4 Conclusion

NRC staff did not reach conclusions based on this section. This information will be provided with the OLA.

### 4.6.5 Instrumentation and Control

#### 4.6.5.1 Introduction

This section states that instrumentation and control for the CRD system is described in PSAR Sections 7.3.1, 7.3.2, and 7.3.3.

#### 4.6.5.2 Regulatory Evaluation

Regulatory evaluation for instrumentation and control systems supporting the CRD system is provided in Chapter 7.

#### 4.6.5.3 Technical Evaluation

Technical evaluation of instrumentation and control systems supporting CRD systems is provided in Chapter 7 of this report.

#### 4.6.5.4 Conclusion

Conclusions concerning instrumentation and control systems supporting CRD systems are provided in Chapter 7 of this report.

### 4.6.6 Control Rod Drive System Description

#### 4.6.6.1 Introduction

PSAR Section 4.6.3 states that the CRD system is composed of three major elements: the electro-hydraulic FMCRD mechanisms, hydraulic control units (HCUs), and the CRD hydraulic subsystem.

The fine motion capability is achieved with a ball-nut and ball-screw arrangement driven by an electric motor. The ball-nut traverses axially as the ball-screw rotates and is keyed to the control rod guide tube to prevent its rotation. A hollow piston rests on the ball-nut, and upward motion of the ball-nut drives this piston and the coupled control rod into the core. The weight of the control rod keeps the hollow piston and the ball-nut in contact during withdrawal. During hydraulic scram, high-pressure water is allowed to flow through the fine-motion control rod drives, lifting the hollow piston from the ball-nut and driving the coupled control rod into the core. A latching mechanism supports the hollow piston following insertion. Additionally, the motor drives the ball-nut to a position just under the hollow piston on any scram signal.

Each HCU furnishes pressurized water for hydraulic scram, on signal from the RPS, to two CRD units (with the exception of the central control rod which is supported by its own HCU). The accumulator provides the stored energy necessary to obtain the required high-pressure, high-flow-rate discharge of water to the two associated FMCRDs in order to fully insert the control rods under any anticipated reactor pressure. The accumulator has a floating piston with nitrogen on one side and water on the other side. The HCU also includes the scram solenoid pilot valve, scram valves, and check valves. Additionally, each HCU provides the capability to adjust purge flow to the drives. A test port is provided on the HCU for connection to a portable test station. to allow controlled venting of the scram insert line to test the FMCRD ball check valve during plant shutdown.

The CRD hydraulic subsystem supplies clean, demineralized water that is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD hydraulic subsystem relies on water from the condensate storage tank (see PSAR Section 9.2.6) or from the condensate and feedwater heating system. During normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs. The CRD hydraulic system is also the source of pressurized water for purging the reactor water cleanup/shutdown cooling system pumps and the nuclear boiler system reactor water-level reference leg instrument lines. Additionally, the CRD hydraulic subsystem provides high-pressure makeup water to the reactor during events in which the feedwater system is unable to maintain reactor water level. Per PSAR Section 15.6.3.10.1, this makeup water is supplied to the reactor via control rod drives.

#### *4.6.6.2 Regulatory Evaluation*

The regulatory evaluation described in Section 4.6.1.2 also applies to this subsection.

#### *4.6.6.3 Technical Evaluation*

Providing a reliable and fail-safe means to rapidly insert control rods using stored hydraulic energy and maintain them in the fully inserted position is an SC1 function of the CRD system. In partial support of this assertion, PSAR Section 4.6.3 describes the design of components involved in achieving hydraulic scram. In each HCU, the scram accumulator stores sufficient energy to fully insert the two assigned control rods at any anticipated reactor pressure. The CRD hydraulic subsystem provides charging water to the HCU accumulators. Check valves in each HCU provide assurance that HCU inventory will be maintained if charging water pressure is lost. A bottle of pressurized nitrogen is attached at the bottom of each HCU accumulator, and a piston separates water from nitrogen within the cylinder. Pressure sensors provide indication of nitrogen pressure locally and in the control room, and loss of nitrogen pressure is alarmed in the control room. Scram accumulators are continuously monitored for leakage, and a level sensor is available in each hydraulic cylinder to indicate leakage of water past the piston. This signal is alarmed in the control room. The PSAR describes instrumentation and alarms which can alert operators if the condition of an HCU accumulator is degraded. PSAR Section 16 indicates that limiting conditions for operation for control rod operability and control rod scram accumulators, including required actions in the event of degraded conditions, will be provided for NRC staff review with the OLA.

PSAR Section 4.6.3.2 describes that a check valve is included with the safety function to close when charging header pressure falls below scram header pressure. It states that a port is included to test this check valve.

Opening of the scram valve in the HCU creates a flow path between the HCU accumulator and the connected fine-motion control rod drives. In normal operation the scram valve is held closed by air pressure applied to its operator. This air pressure is removed by opening of the scram solenoid pilot valve. The scram solenoid pilot valve opens on loss of power from both solenoids. Inadvertent loss of air pressure or loss of power from both pilot valve solenoids would lead to inadvertent scram of the drives associated with a given HCU. PSAR Figure 4.6-8 indicates that pressurized air is supplied to each HCU by the instrument air system through the scram air header of the CRD hydraulic subsystem.

As scram valves are held closed by scram discharge pressure, partial loss of instrument air or the scram header air could lead to partial opening of scram valves and deterioration of HCU pressure and inventory. PSAR Section 4.6 describes instrumentation and control room alarms that may alert operators to loss of HCU pressure before hydraulic scram capability is lost. PSAR Section 16 identifies that technical specifications on control rod scram accumulators will be provided with the OL. Sufficiency of instrumentation and required actions in response to degraded conditions will be reviewed at that time.

NRC staff requested that GVH provide the failure modes and effects analysis (FMEA) for the BWRX-300 CRD system with the similar failure mode and effects analysis performed for the FMCRD systems incorporated in the ESBWR and Advanced Boiling Water Reactor (ABWR) designs. In response to audit question 4.6-5 (ML26091A346), GVH listed the differences between the FMCRD designs of the ABWR, ESBWR, and the BWRX-300 FMCRD designs and identified the effect of the differences on the FMEA. The BWRX-300 FMCRD designs are very similar to the prior approved FMCRD designs. Similar to the ESBWR FMCRD, but unlike the ABWR design, the BWRX-300 FMCRD has stay on latches after scram and uses a Servo motor run-in after scram. Unlike the previous ABWR and ESBWR FMCRD, the BWRX-300 has a fast motor run-in after scram and, in addition to diesel generators for redundant power, the BWRX-300 also has the uninterruptible power supply. GVH stated that the BWRX-300 FMCRD design differences have a negligible effect on the FMEA and that additional features added since the initial ABWR FMCRD design add assurance of rod drop protection. The NRC staff find this assessment reasonable for the CP and thus find that the NRC safety review of the FMCRDs in the ABWR and ESBWR design certification reviews can be leveraged for the review of the BWRX-300 FMCRD review.

An alternative means of opening scram valves is provided by solenoid-operated alternate rod insertion valves on the scram air header. These valves are designed to vent the scram air header on any scram signal, which opens the scram valve in every HCU. Unlike scram solenoid pilot valves, these valves are energized to actuate, and venting through an alternate rod insertion path requires two in-series valves to open. The PSAR states that these valves are not credited in any deterministic or probabilistic safety analysis for BWRX-300.

As described in PSAR Section 4.3.1.2, shutdown margin is calculated assuming failure of the HCU corresponding to the most reactive control rod or rod pair being stuck in the fully withdrawn position. Chapter 15 of this report discusses the effect of a malfunction of a control rod on the safety analysis. PSAR Chapter 4.3.1.2 evaluates the effect of a stuck rod on shutdown margin.

In order to keep control rods inserted following a reduction in flow from the HCU accumulators, the hollow cylinder is designed with latches that engage with slots on the control rod guide tube when the hollow cylinder separates from the ball nut during scram. Hydraulic scram signals also induce FMCRD motors to drive the ball nut to a position just under the hollow cylinder of the control rod.

NRC staff notes that while PSAR Section 4.6 discusses CRDs, the ability of the CRD system to successfully insert control rods is also dependent on measures to ensure integrity of control rods and other core components. NRC staff evaluation of these measures is summarized in Section 4.2 of this report.

NRC staff reviewed the PSAR to understand evaluations that the applicant will perform for consistency with GDC 4. PSAR Section 3.1.1.4 states that SC1 SSCs are designed to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated piping failure accidents including loss-of-coolant accidents. It also specifies that SC2 and SC3 SSCs that mitigate postulated initiating events will be confirmed to be compatible with environmental and dynamic effects present when they are credited. It states that failure of non-SC1 equipment due to environmental and dynamic effects will not impair SC1 equipment.

Although PSAR Section 3.1.1.4 does not clearly identify whether SC1 components are protected from the effects of missiles, PSAR Section 3.5 states that these components will be protected from missiles.

NRC staff confirmed that CRD components needed to perform hydraulic scram are categorized as SC1, seismic category I per PSAR Table 3A-1. This table also identifies that charging or purge water piping above HCUs is categorized as seismic category II, with final categorization pending development of physical layout in the final design, to be provided for review with the OLA (see Table 3A-1 footnotes).

Limitation and Condition 2 of NEDC-33911P-A, Revision 2 notes that NRC staff will review the consequences of pipe failures outside containment resulting, in part, from failures in FMCRD hydraulic lines. NRC staff evaluation of pipe failures outside containment is described in other SER sections.

Based on the review of the system description in the PSAR and provided failure modes and effects analysis, the NRC staff finds that sufficient information concerning the design basis of CRD system components has been provided to support issuance of the CP pursuant to 10 CFR 50.34(a) and 10 CFR 50.35. Review of ability of components to fulfill their design functions following detailed design will be performed as part of the CRN-1 OL review.

NRC staff evaluation of the role of the CRD system in meeting GDC 1, 2, 3, and 4, with instrumentation and control system needed to initiate CRD safety functions, with power required by the CRD system, with operational tests to ensure CRD system performance, with technical specifications that ensure operation of the CRD system supports the CRN-1 safety analysis, and with the quality assurance performed for the CRD system, are provided in Chapters 3, 7, 8, 14, 16, and 17, respectively.

PSAR Section 4.6.3.1 describes an electro-mechanical brake that will be designed to prevent motion of the drive shaft during failure of the pressure retaining parts of the CRD mechanism. PSAR Section 4.6.3.1 also states that the brake will prevent the FMCRD motor from moving the control rod when engaged. The applicant stated that the FMCRD brake is engaged at all times with the exception of when the FMCRD motor is used to drive in/insert control rods and is designed such that if it were to fail, it would fail in the braked position. In the case of the failure of the pressure retaining parts of the drive mechanism, the ESBWR FMCRD was shown to have margin to the FMRD design braking torque requirement (see ESBWR FSER Chapter 4.6 [ADAMS Accession No. ML103470435]). Due to the similarities in the FMCRD designs, the NRC staff preliminarily find the capabilities of the FMCRD brakes to prevent rod ejection to be reasonable if there were a failure of the pressure retaining portions of the FMCRD, but this will be confirmed for the BWRX-300-specfic FMCRDs in the OLA review.

The applicant further explained that if there was a brake failure that resulted in the unintentional release of the brake without the corresponding motor energization and without the failure of the scram line, then the testing showed that the rod drop rate is bounded by the inadvertent single control rod withdrawal at power event analyzed in PSAR Chapter 15.

In PSAR Section 3.1.3.9, TVA proposed alternate PDC 28 in place of GDC 28 and an associated exemption request included in Enclosure 4 of the CPA. The proposed PDC 28 removes rod drop from the list of postulated accidents. In response to RAI 15.2-8 (ML26082A195), the applicant stated that the bounding reactivity insertion DBA is the inadvertent single control rod withdrawal at power event. During the OLA review, the NRC will further evaluate whether the inadvertent single control rod withdrawal at power event is bounding, or if there are other events that have been previously excluded from the design basis because they were bounded by control rod drop but now should be added to the design basis. Further, the staff will evaluate the categorization of events.

The staff's considerations regarding event frequencies for reactivity and power distribution anomalies is located in Section 15.5.5.3 of this SER, including a discussion of the staff's disagreement on the categorization of the single rod withdrawal event as a DBA, and determination that the event must be categorized as an AOO. General considerations for event frequencies are in Section 15.1.3.1 of this report. Additional technical information that should be supplied in the FSAR is identified in Chapter 15 of this report relative to event selection. In addition, the additional justification should be provided in the OL application that the inadvertent single control rod withdrawal at power event is the limiting reactivity-initiated postulated initiating event.

With regard to GDC 28 requirements related to rod drop events, the applicant's exemption request supporting their proposed PDC 28 lists positive means to prevent the control rod drop from occurring and provides an estimate of control rod drop frequency. One method of preventing a control rod drop accident involves detecting separation between the control rod and CRD and initiating a rod block to prevent further withdrawal of the ball nut. The blocked ball nut would restrict the motion of the postulated falling control rod. These separation switches are based on weight and are designed to detect separation at the bayonet coupling between the hollow piston and the control rod drive. Separation between the ball nut and hollow piston is a larger change in weight. The weighing table is within the reactor coolant pressure boundary. A control rod withdrawal block is initiated automatically when separation is detected, which would prevent further withdrawal of the ball nut.

The second means of preventing control rod drop are latches on the hollow cylinder. The PSAR describes that these latches engage with slots in the control rod guide tube when the hollow cylinder departs from the ball nut during a scram.

The third means of preventing a control rod drop discussed in the exemption is the scram follow signal, which causes the FMCRD motor to drive the ball nut to a position just under the control rod drive in the event of a scram signal. A hydraulic scram is a function of the CRD system that is credited in response to most design basis events analyzed in Chapter 15. When there is a hydraulic scram, the hollow cylinder is separated from the ball nut. In lieu of the scram follow function, a failure of the latching mechanism alone would cause a control rod to drop to its pre-scram position.

The fourth means of preventing a control rod drop is the design of the bayonet coupling between the hollow cylinder and the control rod. Disengaging the mechanism requires a 45 degree rotation. The control rod is prevented from completing such a rotation by the adjacent fuel assemblies. There is also an anti-rotation device that engages when the lower component is removed for maintenance and illustrated in PSAR Figure 4.6-10.

The GDC 28 rod drop exemption request also included an estimated frequency of control rod drop based on the credited mechanisms. The applicant noted that the estimate of this failure frequency did not credit action from the operator following observation that the control rod was detached from the drive due to alarms in the control room caused by separation switches or observation of core monitoring instrumentation response. The NRC staff will further evaluate the calculation of the estimated frequency of control rod drop provided during its evaluation in the OL review as well as the FMCRD operating history, but finds the preliminary frequency estimate reasonable for the PSAR because it is similar to that calculated for the similarly designed ESBWR FMCRD (ML071930199), which the NRC stated was reasonable in Chapter 4.6 of the ESBWR FSER (ML103470435). In the OL application, the applicant should provide and justify details of the calculation of the frequency of a control rod drop accident (CRDA).

The preliminary staff review of this exemption considered the adequacy of the principal design criterion and the design functions described for each component. Adequacy of components to fulfill these design functions consistent with the principal design criterion will be reviewed with the OLA. Overall, the NRC staff finds PDC 28 sufficient to support issuance of the CP based on the design features of the BWRX-300 FMCRDs, specifically the features intended to prevent a control rod drop. As a part of the evaluation of PDC 28 in the OLA review, the NRC will evaluate the CRDA beyond-design basis event analysis results in the probabilistic risk assessment consistent with limitation and Condition 5.2 of NEDC-33912P-A, Revision 2, “BWRX-300 Reactivity Control” (ADAMS Package Accession No. ML23048A015). As such, in the OL application, the applicant should provide CRDA beyond-design basis event analysis results in the probabilistic risk assessment must be provided, consistent with limitation and Condition 5.2 of NEDC-33912P-A, Revision 2.

The staff did not perform a detailed review of the exemption request for the CP review, because additional information is required to support staff findings on the exemption request, as specified in the preceding OL action items. The staff expects to make a finding relative to the exemption request once the additional information is received, to support a finding relative to the OL.

PSAR Section 4.6.3.3.2 states that the CRD pumps have a function to provide makeup in the event of a LOCA that cannot be isolated. Additionally, the PSAR Chapter 3 discussion of GDC 33 indicates that the CRD hydraulic subsystem is credited in providing CRD makeup and in unisolable leak events. Specifically, the CRD makeup function is credited for meeting GDC 33, yet is categorized as SC3 and non-seismically qualified. The NRC staff considers systems that satisfy the reactor coolant makeup function required by GDC 33 to be important-to-safety and thus subject to GDC 1, 2, and 4. The applicant provided justification for the classification of these components and described how GDC 1, 2 and 4 are met for the CRD hydraulic subsystem in response to audit question 4.6-2 (ML26061A096), however the NRC staff could not reach a finding regarding the CRD hydraulic subsystem SSC classification based on the information the applicant provided. Based on the location of the CRD hydraulic subsystem components, which are mostly near SC1 and SC2 seismically qualified SSCs, the NRC staff believe that there would be some assurance that the CRD hydraulic subsystem components would be protected from impacts resulting from natural phenomena and dynamic effects that may result from equipment failures, such as missiles, pipe whipping, and discharging fluids. However, the applicant will need to provide additional justification for the SSC classification of the CRD hydraulic subsystem during the review of the OL, including demonstration that unislolated leaks will not uncover the core. As such, in support of the OL application, the applicant should provide in the FSAR additional justification for the SSC classification of the CRD hydraulic subsystem and how GDC 1, 2, and 4 are met for the subsystem components.

In PSAR Section 3.1.3.7, TVA proposed PDC 26 that states that two independent reactivity control systems of different design principles shall be provided. The first uses two diverse means of inserting control rods to assure that under conditions of normal operation, including the effects of AOOs, and with appropriate margin to stuck rods, SAFDLs are not exceeded. The second system shall be capable of inserting negative reactivity to assure a reactor shutdown from full power operating conditions if both diverse means of inserting control rods were to fail. Both of the reactivity control systems shall be capable of holding the reactor core subcritical under cold conditions. TVA states that they will meet PDC 26 with the FMCRDs, which can be inserted by two diverse means: the HCU and the motor-driven run-in function, as the first system and the BIS as the second independent and diverse system to meet PDC 26. More details of the BIS are included in Chapter 9.3.10 of the PSAR. The capabilities of the BIS will be further evaluated in the OLA review. Discussion of other PDC 26 requirements on reactivity control systems is provided in Section 4.3 and Section 15 of this report.

The NRC staff find TVA's proposed PDC 26 and description of how it will be met acceptable because it meets the underlying intent of GDC 26 to provide two diverse and independent means of controlling reactivity. The NRC staff defines two systems to be "independent and diverse" if there are no shared systems or components and they are of a design which is different enough such that no common failure modes exist (see RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors" [ADAMS Accession No. ML17325A611]). The FMCRDs and the BIS are two diverse and independent means of controlling reactivity.

#### *4.6.6.4 Conclusion*

The NRC staff finds preliminary CRD design basis to be sufficiently consistent with regulatory criteria to support issuance of the CP. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

### 4.6.7 Conclusion

The staff has reviewed the design basis of the CRDS to confirm that it will support meeting regulatory requirements.

The review has determined the adequacy of the applicant's proposed design criteria, design basis functions, and safety classification of the CRDS and the requirements for providing a safe shutdown during normal operation, anticipated operational occurrences, and accident conditions, including single failures. The staff concludes that the preliminary design basis of the CRDS provided in PSAR Section 4.6 is consistent with the requirements of GDC 4, 23, 25, 27; PDC 26, 28, 29; and 10 CFR 50.62(c)(3). Therefore, the NRC staff finds that the information in PSAR Section 4.6 adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

As described in the staff's safety evaluation above, the FSAR submitted with the OLA must include information to satisfy the OL action items noted previously in this section.

## 4A Thermal-Hydraulic Stability

### 4A.1 Summary of Stability Characteristics

Appendix 4A of the CRN-1 PSAR discusses the thermal-hydraulic stability characteristics of the BWRX-300 reactor core. The appendix explains the mechanisms by which the BWRX-300 reactor can be susceptible to coupled neutronic and thermal-hydraulic oscillations due to density wave oscillations and describes how core design features and analysis methods are used to demonstrate adequate stability margins. The applicant identifies the most limiting instability condition as core-wide oscillations and states that a conservative design criterion is imposed on the core decay ratio under all conditions of normal operation and anticipated operational occurrences. The PSAR describes the use of the proposed TRACG code and methodology for stability analysis documented in NEDC-34270, "BWRX-300 Stability Analysis," licensing topical report, with a decay ratio acceptance criterion of ≤0.80. Design features addressed for stability include the small core size, reliance on natural circulation, a tall chimney to enhance flow and damping, and optimized core orifice design. The appendix also describes the analytical methods used including the use of an implicit integration scheme for the stability analysis in the time domain.

### 4A.2 Regulatory Evaluation

This section summarizes the regulatory requirements and guidance that provide the acceptance criteria for NRC staff review of the BWRX-300 thermal-hydraulic stability analysis and design.

**Applicable GDC from 10 CFR Part 50, Appendix A:**

**GDC 10 (Reactor Design):**
Requires that the core and associated coolant, control, and protection systems are designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

**GDC 12 (Suppression of Reactor Power Oscillations):**
Requires that power oscillations which could result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

**Applicable Federal Regulations:**

**10 CFR 50.34 (Contents of Applications; Technical Information):**
Requires that PSARs include information demonstrating that the design and analyses of the reactor and safety systems are sufficient to provide reasonable assurance of adequate protection of public health and safety.

**Applicable Regulatory Guidance:**

**NUREG-0800, SRP Section 4.4 (Thermal and Hydraulic Design):**
Provides acceptance criteria for the thermal-hydraulic design of the reactor core, including requirements for demonstrating stability against power oscillations and methods for analyzing single-phase and two-phase fluid flow

**NUREG-0800, SRP Section 15.9 (BWR Stability):**
Describes staff review practices and criteria for BWR stability, including the use of analytical codes, decay ratio acceptance criteria, and required demonstration of margin under all operating conditions.

Other relevant regulations, such as GDC 13, 29, GL 94-02 are no longer relevant to this section, as a detect and suppress function is not used in the BWRX-300 design and the applicant's preliminary analysis is intended to demonstrate that the reactor would not reach instability prior to a reactor scram, as the result of the protection system activation.

## 4A.3 Technical Evaluation

### *4A.3.1 Introduction*

A natural circulation BWR is potentially subject to instabilities in the form of core-wide, regional and channel oscillations due to density wave oscillations for high power operation as defined as Type 2 instability or Type 1 instability during start up with low system pressure.

The staff considered several key features of the BWRX-300 reactor in evaluating its design with respect to stability. The BWRX-300 reactor as proposed for CRN-1 is designed to have a small core with only 240 fuel bundles. As it will be loaded with full length GNF2 fuel, it has a much smaller core radius and longer active fuel length than the GVH ESBWR design, which the staff previously evaluated with respect to stability. Tight neutronic coupling reduces the likelihood of regional mode oscillations. In addition, the tall chimney and optimized inlet core orifice design reduces the likelihood of the thermal-hydraulic instabilities. The staff audited calculational results and other documentation as part of its review (ML25191A113).

The PSAR references NEDC-33912-P-A, Revision 2, "BWRX-300 Reactivity Control," and states that compliance to GDC 12 was previously addressed in that topical report. However, GDC 12 compliance is not confirmed through that topical report, but through the staff's review and evaluation of the stability demonstration at the PSAR and FSAR stages of licensing. Limitation and Condition (L&C) 5.3 of NEDC-33912-P-A, Revision 2, specified such a demonstration must be made. Rather than satisfying all aspects of L&C 5.3, the staff reviewed the PSAR to determine whether the methodology and results support a preliminary determination relative to GDC 12 that sufficient information exists to support issuance of a CP.

### *4A.3.2 Stability Analysis Methodology*

The applicant uses the TRACG coupled thermal-hydraulics and three dimensional neutronics code to analyze stability margins. TRACG is a time-dependent code with a full two-fluid representation. NEDE-33083P, Supplement 1, documents the TRACG04 code and analysis methodology for calculating stability margins in the GVH ESBWR certified design and the corresponding SER presents the staff's approval. Section 4A.1 of the PSAR cites NEDC-33922-P-A, "BWRX-300 Containment Evaluation Method," as an NRC approval of its TRACG code for BWRX-300. The staff notes that this approval was limited to its use for containment evaluations and was not an approval for its use in stability analysis. For BWRX-300, the approval to use TRACG to perform stability analyses is in NEDC-34270, Revision 1 "BWRX-300 Stability Analysis," (ML26089A383) in which GVH activates the implicit integration scheme to more precisely determine the decay ratio margin.

As part of the review of NEDC-34270, Revision 1, the staff independently performed audit calculations of both the core wide and regional oscillation decay ratio under nominal rated operating conditions using the TRACG code and the BWRX-300 input model. The input model mapped the entire core to 240 thermal-hydraulic CHANNEL components achieving the one-to-one neutronic node to thermal-hydraulic node mapping. The audit calculation independently confirmed the decay ratio results provided in the NEDC-34270, Revision 1 LTR.

The staff has determined that the stability method's implementation in the CRN-1 PSAR is appropriate as a preliminary methodology, but in support of the OL application, the FSAR must demonstrate that the limitations and conditions specified in the approved version of NEDC-34270, Revision 1, are met.

Staff find this method of referencing nuclear analysis codes, methods and analytical results sufficient for the purposes of the CPA. In order to ensure that the licensing basis is clearly established and can be maintained during operation of the facility, the FSAR provided with the OLA should include a listing of the stability analysis codes, methods and analytical results (within either technical or topical reports) that are either incorporated by reference, or are described, outlined or summarized in sufficient detail to be considered described within FSAR for the purpose of applying 10 CFR 50.59, as described in Section 3.10 of NEI 96-07, Revision 1, as endorsed by RG 1.187.

*4A.3.3 Thermal-hydraulic Instabilities and Stability Margin*

Based on the design features of the BWRX-300 and using the stability method described above, the applicant analyzed the stability margin against Type 2 density wave oscillations under nominal rated conditions using the equilibrium core design described in PSAR Section 4.3 and reported the results in Section 15.5.2 of the PSAR. The staff's evaluation of these results is in Section 15.5.2 of this SER.

*4A.3.4*

PSAR Section 4.4 includes Table 4.4-2, specifying the expected BOC core flow and nominal feedwater temperature vs. power. This relationship is central to the underlying assumptions in the stability analyses, therefore at the OL phase, the applicant should address the following:

> Pursuant to 10 CFR 50.36(c)(2)(ii)(b), the applicant must include technical specification controls within its OLA to limit normal operation to within the initial conditions assumed for the design basis stability analyses.
>
> The OLA must include a power ascension test to confirm that actual plant operation is consistent with, or conservative relative to, the initial conditions for feedwater temperature vs. power used for the design basis stability calculations, with provisions for reanalysis if expected operation is not consistent with, or not conservative relative to, the analyses.

During start up, when the system pressure is low, two potential non-density wave mechanisms for flow oscillations at low pressure for a natural circulation BWR may exist. The first is a "geysering" flow oscillation, which results from vapor flashing at the top of the chimney region because the saturation temperature is lower at the chimney top than at the core because of the pressure difference. As vapor flashing starts, core flow is increased and the core exit enthalpy is reduced, which stops the vapor generation, and a flow oscillation may occur. The other non-density wave flow oscillation is the "Type 1" instability. These oscillations occur when there is voiding in the chimney, which leads to a reduction in the hydrostatic head in the chimney and an increase in flow. Oscillations of this kind are inevitable in a natural circulation reactor because this instability region must be crossed before a steady two-phase voided region is established in the chimney. As it is stated in NEDC-34270, Revision 1, "BWRX-300 Stability Analysis," BWRX-300 will experience thermal hydraulic instabilities during start-up. However, there is no information in either Appendix 4A or Section 15.5.2 to show the oscillation amplitude, nor the CHF margin, therefore at the OL phase, the FSAR should provide analytical results justifying compliance with GDC 10 and 12 during startup conditions

The PSAR does not identify the actual secondary side and control system design. The uncertainty introduced by these systems must be considered within the stability analysis, therefore at the OL phase, the FSAR must justify that the initial conditions to be assumed for final design basis stability analyses, including those performed at off-rated conditions, account for the uncertainties in measurements, manufacturing and installation and secondary side control system response.

*4A.3.5 Instabilities During Anticipated Transients Without Scram*

During the review of PSAR Appendix 4A, the staff noted that neither this appendix nor PSAR Section 15.5.2 contain any instability analysis for ATWS events. No rationale for its exclusion was provided. SRP Section 15.9 states “Thermal-hydraulic stability performance of the core during an ATWS event should not exceed acceptable fuel design limits.” Without ATWS analyses, including thermal hydraulic instability analysis, there is no information for staff to confirm acceptable fuel design limits are not exceeded during an ATWS event. The resolution of this issue is further discussed in Section 15.5.5 of this report. The staff notes that in accordance with 10 CFR 50.62(d), ATWS is addressed at the FSAR stage. The staff notes that FSAR should include a summary of ATWS events’ instability analyses.

**4A.4 Conclusion**

The staff confirmed that the thermal-hydraulic stability characteristics of the BWRX-300 reactor core described in this appendix are enhanced by design features such as a small core size, reliance on natural circulation, a tall chimney to enhance flow and damping, and an optimized core orifice design. The preliminary stability analysis results summarized in Section 15.5.2 indicate a core wide oscillation decay ratio less than the acceptance criteria of 0.8 under nominal rated operating conditions with an equilibrium core design.

The staff recognizes that Appendix 4A and Section 15.5.2 are preliminary and the staff cannot make a final determination on compliance with GDC 10 and GDC 12. However, the staff finds that the applicant has identified relevant requirements and provided sufficient description of preliminary facility design and criteria. The applicant has identified appropriate regulatory requirements and guidance and a methodology to perform final analyses at the FSAR stage. The staff finds this level of preliminary information sufficient to support issuance of the CP pursuant to the regulations of 10 CFR 50.34(a) and 10 CFR 50.35, as applicable.

As described in the staff’s evaluation above, and in addition discussions related to stability that are identified in Section 15.5.2 of this report, the FSAR submitted with the OLA should include the additional technical information identified previously in this section.

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