REACTOR

4.1 Introduction

Chapter 4, "Reactor," of the Economic Simplified Boiling-Water Reactor (ESBWR) Design Control Document (DCD), Tier 2, Revision 7, describes the mechanical components of the ESBWR reactor and reactor core, including the fuel system design (fuel rods and fuel assemblies), nuclear design, thermal-hydraulic design, reactor materials, and functional design of the control rod drive (CRD) system.

DCD Tier 2, Chapter 4, also identifies certain areas as "Tier 2*" information, departures from which require prior approval from the staff of the U.S. Nuclear Regulatory Commission (NRC). Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Appendix A, "Design Certification Rule for the U.S. Advanced Boiling Water Reactor," Section VIII.B(6.a), provides a definition and the criteria governing Tier 2* information.

The following sections in DCD Tier 2, Chapter 4, include Tier 2* information:

- Section 4.2.7
- Section 4.3.6
- Section 4.4.8
- Appendix 4A
- Appendix 4B
- Appendix 4C

4.2 Fuel System Design

The fuel system comprises the fuel assembly and the reactivity control assembly. The fuel assembly consists of the full-length and part-length fuel rods, grid spacers, water rods, upper and lower tie plates, and the channel. Appendix 4B, "Fuel Licensing Acceptance Criteria," to the ESBWR DCD defines the fuel assembly design criteria that must be satisfied by any fuel design to be loaded into the ESBWR core. Appendix 4C, "Control Rod Licensing Acceptance Criteria," to the ESBWR DCD defines the control blade design criteria that must be satisfied for any control blade design used in the ESBWR core.

4.2.1 Regulatory Criteria

The NRC staff reviewed DCD Tier 2, Section 4.2, "Fuel System Design," Appendix 4B, "Fuel Licensing Acceptance Criteria" and Appendix 4C "Control Rod Licensing Acceptance Criteria," in accordance with the regulatory guidance for the review of fuel system design, including adherence to applicable general design criteria (GDC) discussed in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), Section 4.2, "Fuel System Design," Revision 2, issued July 1981. The staff performed a comparison of the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, generic issues (GIs), bulletins (BLs), generic letters (GLs), or technically significant acceptance criteria (except Appendix 4B, Interim

Criteria and Guidance for the reactivity initiated accidents) beyond those identified in the version used by the staff. Therefore, the staff finds that the use of draft Revision 3 of SRP Section 4.2, issued in July 1982, is acceptable for this review.

The following GDCs (10 CFR 50, Appendix A) are applicable in SRP Section 4.2:

- GDC 10, "Reactor Design," as it provides assurance that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs);

- GDC 27, "Combined Reactivity Control Systems Capability," as it relates to the combined effect of the reactivity control system being designed with appropriate margin and capability (i.e., in conjunction with poison addition to reliably control reactivity changes including accident conditions and margin to account for stuck rod(s)) to cool the core; and,

- GDC 35, "Emergency Core Cooling," as it relates to emergency core cooling so that following any loss of reactor coolant, 1) fuel and clad damage that could interfere with core cooling is prevented, and 2) clad metal-water reaction is limited to negligible amounts.

In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance of the following:

- The fuel system is not damaged as a result of normal operation and AOOs.

- Fuel system damage is never so severe as to prevent control rod insertion when it is required.

- The number of fuel rod failures is not underestimated for postulated accidents.

- Coolability is always maintained.

The staff reviewed the Tier 1, Tier 2, and Tier 2* fuel design and control blade design acceptance criteria to ensure that the requirements outlined in SRP Section 4.2 are satisfied.

The DCD requirements for the contents of applications appear in 10 CFR 52.47. SRP Section 14.3.4 provides guidance related to the approval status of fuel system design and the designation of DCD requirements, including the following:

- The specific fuel, control rod, and core designs presented in Tier 2 will constitute an approved design that may be used for the combined operating license (COL) first-cycle core loading without further NRC staff review. If any other core design is requested for the first cycle, the COL applicant or licensee must submit for staff review the specific fuel, control rod, and core design analyses as described in DCD Tier 2, Chapters 4, 6, and 15. Much of the detailed supporting information in Tier 2 for the nuclear fuel, fuel channel, and control rod, if considered for change by a COL applicant or licensee referencing the certified standard design, would require prior NRC approval. Therefore, for the evolutionary designs, the staff concluded that this information should be designated as Tier 2* information. However, the staff allowed some of the Tier 2* designation to expire after the first full-power operation of the facility, when the detailed design would be complete and the core performance characteristics would be known from the startup and power ascension test programs. The NRC bears the final responsibility for designating which material in Tier 2 is Tier 2*.

- Inspections, tests, analyses, and acceptance criteria (ITAAC) are not required for Tier 1 information in the fuel, control rod, and core design areas because of the requirement for prior NRC approval of any proposed changes to the approved design.

- Post-fuel-load testing programs (e.g., startup testing and power ascension testing) verify that the actual core performs in accordance with the analyzed core design.

Only fuel assembly and control blade designs that satisfy all of the ESBWR design requirements and have been reviewed and approved by the NRC are to be used during the initial core (Cycle 1) in any facility that adopts the ESBWR certified design.

4.2.2 Summary of Technical Information

DCD Tier 2, Section 4.2.1.1, describes the design basis of the ESBWR fuel assembly. The thermal-mechanical fuel design provides the following capabilities:

- substantial fission product retention capability during all potential operational modes to comply with 10 CFR Part 20, “Standards for Protection against Radiation,” 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” and 10 CFR Part 100, “Reactor Site Criteria”

- sufficient structural integrity to prevent operational impairment of any reactor safety equipment

The fuel assembly and its components are designed to withstand the following:

- predicted thermal, pressure, and mechanical interaction loadings occurring during startup testing, normal operation, and AOOs, infrequent events accidents, and mechanical loads from seismic events

- loadings predicted to occur during handling

In DCD Tier 2, Section 4.2 and Appendix 4B provide the design requirements for the fuel design, along with a brief description of the design evaluations. An earlier version of DCD (Tier 1, Section 2.8) provided principal fuel design and performance requirements. In the final DCD, these criteria were reclassified as Tier 2* and moved to Appendix 4B.

DCD Tier 2, Section 4.2.1.2, describes the design basis of the ESBWR control blades. These structures are designed to have the following capabilities:

- sufficient mechanical strength to prevent displacement of their reactivity control material

- sufficient mechanical strength to prevent deformation that could inhibit their motion

In DCD Tier 2, Section 4.2 and Appendix 4C provide the design requirements for the ESBWR control blades, along with a brief description of the design evaluations. An earlier version of DCD Tier 1, Section 2.9, provided the principal control blade design and performance requirements. In the final DCD, these criteria were reclassified as Tier 2* and moved to Appendix 4C.

4.2.3 Staff Evaluation

The regulatory criteria and the specific fuel, control rod, and core designs presented in Tier 2 will constitute an approved design that may be used for the COL first-cycle core loading without further NRC staff review. An approved fuel design with specific design and performance requirements is a foundation for determining the acceptability of the plant systems' response to AOOs and postulated accidents.

To fulfill these regulatory requirements, the ESBWR DCD references the approved GE14E fuel assembly design documented in the following licensing topical reports (LTRs): NEDC-33240P, Revision 1, "GE14E Fuel Assembly Mechanical Design Report," and NEDO-33242P Revision 2, "GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report." The approved Marathon control blade design is documented in topical report NEDO-33243P, Revision 2, "ESBWR Control Rod Nuclear Design Report," and NEDE-33244P, Revision 1, "ESBWR Marathon Control Rod Mechanical Design Report." The staff documented the basis for its approval of the GE14E fuel design and the Marathon control rod design in the safety evaluations for LTRs NEDC-33240P, Revision 1; NEDC-33242P, Revision 2; NEDE-33243P, Revision 2; and NEDO-33244, Revision 1. The safety evaluation report (SER) for NEDC-33326P Revision 1, "GE14E for ESBWR Initial Core Nuclear Design Report," provides the staff evaluation of the initial core fuel design and core loading pattern.

During the July 2007 General Electric Hitachi Nuclear America, LLC (GEH) control blade and fuel assembly design audit, the staff found that the mechanical design of the ESBWR Marathon control rod blade differed from that presented in NEDE-33243P and NEDE-33244P. The staff requested GEH to issue a revision to these reports that would document the revised design of the ESBWR Marathon and also capture any applicable responses to requests for additional information (RAIs) from the staff's review of the Marathon-5S control blade design for use in operating reactors (RAI 4.9-12). GEH responded to RAI 4.9-12 by noting that it had addressed the differences in NEDE-33243P, Revision 2, and NEDE-33244P, Revision 1. The staff reviewed the LTRs which showed that the differences were addressed, therefore; RAI 4.9-12 was resolved.

4.2.3.1 ESBWR DCD Tier 1

The applicant has reclassified the ESBWR fuel and control blade Tier 1 principal design and performance requirements originally specified in Tier 1, Sections 2.8 and 2.9 (in DCD Revision 3), and moved them to Tier 2, Appendices 4B and 4C (respectively) (See Section 4.2.3.2 below).

Even though the applicant deleted the ITAAC for fuel and control rod (Tier 1, Sections 2.8 and 2.9) in later revisions of the DCD, the following fuel-related design commitments are included in the ITAAC for the reactor pressure vessel (RPV) system (Table 2.1.1-3) for verification:

(1) The initial fuel to be loaded into the core will withstand flow-induced vibration and maintain fuel cladding integrity during operation.

(2) The fuel bundles and control rods for initial core have been fabricated in accordance with the approved fuel and control rod design.

(3) The reactor internals arrangement will conform to the fuel bundle, instrumentation, neutron sources, and control rod locations shown in Tier 1, Figure 2.1.1-2.

In addition, the ITAAC for the nuclear boiler system (Table 2.1.2-3) include the following design commitments:

(1) The pressure loss coefficient of each of the following components is within the uncertainty band of the pressure loss coefficient used in the natural circulation flow analysis:

- steam separator
- fuel bundle
- fuel support piece orifice
- control rod guide tubes
- shroud support

The hydraulic diameter, the geometry of heated surfaces, and flow area in fuel assemblies are within the uncertainty band of the geometry used in the natural circulation flow analysis.

During the review of the GE14E fuel assembly design, the staff issued RAI 4.8-7 to request an explanation about the lack of mechanical testing for flow-induced vibration for the proposed bundle design. In their response GEH proposed specific flow-induced vibration testing for the design of any fuel assembly to be loaded in the ESBWR. DCD Tier 1, Section 2.1.1, Table 2.1.1-3, lists the required testing. NEDC-33240P identifies the acceptance criteria specific to the GE14E fuel design therefore, based on the applicant's response, RAI 4.8-7 was resolved.

4.2.3.2.1 ESBWR DCD Tier 2

The fuel system has been defined as consisting of the fuel assembly and the reactivity control assembly. The fuel assembly comprises the fuel bundle, channel, and channel fastener. The fuel bundle comprises full-length and part-length fuel rods (some of which may contain burnable neutron absorbers), water rods, spacers, springs, and assembly fittings. Appendix 4B to the DCD contains a set of design criteria to be satisfied by new fuel designs to be loaded into an ESBWR reactor.

A previous version of DCD Tier 2, Section 4.2.1.1.4, stated that the cladding oxide thickness itself is not separately limiting, and therefore, no design limit on cladding oxide thickness is specified. Likewise, a previous version of DCD Tier 2, Section 4.2.1.1.5, stated, “Mechanical properties testing demonstrates that the cladding mechanical properties are negligibly affected for hydrogen contents far in excess of that experienced during normal operation.” The staff was concerned that these statements were too general and needed to be supported by mechanical testing data. The staff issued RAIs 4.2-2 and 4.2-4 to request that corrosion limits, both oxide thickness in microns and hydrogen content in parts per million, be quantified for each fuel rod design. At a minimum, the basis of these design limits should include: (1) an oxide thickness that has been specifically accounted for in mechanical design calculations and limits localized surface defects that may promote nonuniform mechanical properties and (2) a hydrogen content that maintains the cladding strain design limit (e.g., 1.0-percent plastic plus elastic strain).

After several supplements regarding the original RAI requests, GEH proposed specific corrosion limits that support the fuel mechanical design and cladding strain criterion for the GE14E fuel design. Section 3.2 of the SER for NEDC-33240P and NEDC-33242P documents the basis for NRC’s approval of the corrosion limits for GE14E and therefore, RAIs 4.2-2 and 4.2-4 were resolved.

DCD Tier 2, Section 4.2.3.1, refers to the GSTRM (GSTR—Mechanical Fuel Model) topical report NEDC-31959P, “Fuel Rod Thermal Analysis Methodology (GSTRM),” issued April 1991, as the approved fuel rod thermal-mechanical design model. The staff issued RAI 4.2-3 requesting the licensing history of GSTRM, including the staff’s review and any subsequent changes to the various fuel performance models within GSTRM. In their response the applicant provided documentation on GSTRM and identified several code modifications. In addition, the applicant updated the cited GSTRM report in the DCD. Based on the applicant’s response and the documentation they provided regarding code modifications, RAI 4.2-3 was resolved.

While performing FRAPCON-3 benchmark calculations in support of the GE14E fuel assembly design topical report, the staff identified a potential nonconservatism in the GSTRM fuel temperature calculation. It is believed that the lack of a burnup-dependent uranium oxide (UO_2) thermal conductivity model is responsible for differences observed between identical FRAPCON-3 and GSTRM calculations. A nonconservative fuel temperature prediction would impact several thermal-mechanical design analyses (e.g., fuel melt, fission gas release) and subsequently, the input to safety analyses (e.g., loss-of-coolant accident (LOCA) stored energy, gap conductivity). The staff accepts the use of the GSTRM model for both gap conductance and thermal conductivity in the ESBWR design certification. The conclusions and limitations for ESBWR TRACG LOCA analyses contained in the NRC staff evaluation of GEH’s Part 21 report (Appendix F to the safety evaluation for NEDC-33173P) are applicable to this safety evaluation. The NRC must approve the use of other methods or analysis strategies for the ESBW. Details of staff evaluation of this issue are included in Section 21.6.3.2.14 of this report.

A previous version of DCD Tier 2, Section 4.2.4.9, stated, "Subsequent Marathon designs or absorber section loadings will be within ±5 percent Δk/k of the initial ESBWR Marathon design." The staff issued RAI 4.2-9 to request clarification of the meaning and the intent of this sentence. In its response regarding the proposed requirement and change criteria, GEH agreed to remove any implied change process and to revise the DCD text accordingly. The finds the revised text in DCD Section 4.2.4.9 acceptable, therefore, RAI 4.2-9 was resolved.

The summary of the changes made in the fuel topical reports were reviewed by the staff and as documented in Section 3.2 of the SER for NEDC-33240P and NEDC-33242, the NRC approved the GE14E fuel design up to the specified rod power envelopes subject to the limitations in the NEDC-33240P and NEDC-33242P SER.

4.2.3.2.2 Appendix 4B Fuel Licensing Acceptance Criteria

The original text of Appendix 4B was modeled after GESTAR-II and appeared to be an overview of a fuel design change process. Appendix 4B to DCD Tier 2 should define the specific Tier 2 and Tier 2* thermal and mechanical fuel design and performance requirements. A separate fuel assembly mechanical design topical report (or a COL application) would then address these requirements to demonstrate, using approved models and methods, the acceptability of a proposed fuel assembly design for the ESBWR. The design certification process requires that the NRC specifically review and approve the fuel assembly design employed in the initial core (Cycle 1) in any facility that adopts the ESBWR certified design.

In its response to RAI 4.2-5 regarding the documented change process, the applicant stated that it would revise Appendix 4B to remove all of the design process information and provided a significantly revised version. Staff concerns with the proposed revision include the lack of specific Tier 2 and Tier 2* fuel thermal and mechanical design requirements and the continued inclusion of a critical power correlation change process. In response to RAI 4.2-5, Supplements 1–3, GEH defined specific thermal-mechanical design and performance requirements and removed the description referring to the change process mentioned in the preceding paragraph. DCD Revision 4 incorporates these changes. Based on the applicant's response, RAI 4.2-5 was resolved.

Principal Fuel Design and Performance Requirements

DCD Tier 2, Appendix 4B.1, states that the specific fuel design to be used in any facility that adopts the ESBWR certified design must comply with the following fuel design and performance requirements:

(1) Fuel rod failure is predicted not to occur as a result of normal operation and AOOs.

(2) Control rod insertion will not be prevented as a result of normal operation, AOOs, or postulated accidents.

(3) The number of fuel rod failures will not be underestimated for postulated accidents.

(4) Coolability will be maintained for all design-basis events, including seismic and LOCA events.

(5) SAFDLs (thermal and mechanical design limits) will not be exceeded during any condition of normal operation, including the effects of AOOs.

(6) In the power operating ranges, the prompt inherent nuclear feedback characteristics will tend to compensate for a rapid increase in reactivity.

(7) The reactor core and associated coolant, control, and protection systems will be designed to ensure that power oscillations that can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

As a result of the reclassification of these design and performance requirements (after DCD Revision 3), the following text and RAI responses may refer to DCD Tier 1, Section 2.8, as opposed to the requirements' final location as Tier 2* criteria in Appendix 4B.

DCD Tier 1, Revision 1, Section 2.8 defined six principal requirements. In RAI 4.2-13, the staff requested clarification on whether these six requirements are, in fact, Tier 1 fuel design requirements. In its response to RAI 4.2-13 regarding the fuel design requirements, the applicant stated that the advanced boiling-water reactor (ABWR) DCD Tier 1 design requirements were more appropriate than those originally defined for the ESBWR (in DCD Tier 1, Revision 1). As a result, the fuel design requirements were modified (as shown above). DCD Revision 3 fuel design requirements (1) through (5) conform to the regulatory criteria specified in Section 4.2.1. Therefore, based on the applicant's response, RAI 4.2-13 was resolved.

Fuel design requirement (6) complies with GDC 11, "Reactor Inherent Protection," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. Fuel design requirement (7) complies with GDC 12, "Suppression of Reactor Power Oscillations." Based on consistency with past certified designs and compliance with current regulatory criteria, the staff finds the fuel design requirements acceptable.

Fuel Thermal-Mechanical Design Requirements

The revised Tier 2* fuel thermal-mechanical design requirements provided in RAI 4.2-5, Supplement 1, are also listed below:

(1) The cladding creepout rate due to fuel rod internal pressure shall not exceed the fuel pellet irradiation swelling rate.

(2) The maximum fuel center temperature shall remain below the fuel melting point.

(3) The cladding circumferential plastic strain during an AOO shall not exceed 1.00 percent.

(4) The fuel rod cladding fatigue life usage shall not exceed the material fatigue capability.

(5) Cladding structural instability, as evidenced by rapid ovality changes, shall not occur.

(6) Cladding effective stresses/strains shall not exceed the failure stress/strain.

(7) The as-fabricated fuel pellet evolved hydrogen at greater than 1,800 degrees Celsius (C) (3,272 degrees Fahrenheit (F) shall not exceed prescribed limits.

With the exception of the fuel melt design limit (requirement (2)) and cladding strain design limit (requirement (3)), the revised Tier 2* fuel design requirements are consistent with currently approved fuel design criteria and are acceptable.

With respect to fuel melting, the staff had concerns about allowing limited fuel melting during an AOO and the definition of core-wide versus local events. In a previous version of DCD Tier 2, Appendix 4B.2 stated, "For local AOOs such as rod withdrawal error, a small amount of calculated fuel pellet centerline melting may occur, but is limited by the 1 percent cladding circumferential plastic strain criterion." In RAI 4.2-6 the staff expressed concerns with: (1) the ability to accurately model fuel volumetric expansion as fuel enthalpy approached incipient melt temperatures and (2) the ability to accurately model the involved fuel pellets in future operation. In its response to RAI 4.2-6 the applicant stated that the rod withdrawal error during refueling has been classified as an infrequent event and that it would remove the statement regarding fuel pellet melting and revise the DCD accordingly. Based on the applicant's response, RAI 4.2-6 was resolved. Chapter 15 of this report discusses and resolves the reclassification of Chapter 15 events, which was an open item in RAI 15.0-15.

Furthermore, the staff would not accept fuel melting for any AOO or infrequent event. On a related subject, the interim criterion for reactivity-initiated accidents (e.g., control rod drop) precludes fuel melting in order to meet the requirements of GDC 28, "Reactivity Limits."

In the revised Appendix 4B Tier 2* fuel design requirements (DCD Tier 2, Revision 3), the text states, "…fuel melting during normal steady-state operation and whole core anticipated operational occurrences are not expected to occur." This statement implies that it is acceptable to experience fuel melt during local AOOs. In Revision 6 of DCD Appendix 4B, Section 4B.2, item (2), "Fuel temperature," the above statement was revised to state, "…fuel melting, during normal steady-state operation and anticipated operational occurrences does not occur." The staff finds this acceptable because it satisfies the Section 4B criteria.

<u>Nuclear Design Requirements</u>

The revised Tier 2* nuclear design requirements are listed in DCD Appendix 4B and are listed below:

1) A negative Doppler reactivity coefficient is maintained for any operating condition.

2) A negative core moderator void reactivity coefficient resulting from boiling in the active flowchannels is maintained for any operating conditions.

3) A negative moderator temperature reactivity coefficient is maintained for temperatures equal or greater than hot shutdown.

4) To prevent a super prompt critical reactivity insertion accident originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel is negative.

5) A negative power reactivity coefficient (as determined by calculating the reactivity change due to an incremental power change from a steady-state base power level) is maintained for all operating power levels above hot shutdown.

6) The core is capable of being made subcritical with margin in the most reactive condition throughout an operating cycle with the most reactive control rod, or rod pair, in the full-out position and all other rods fully inserted.

The six Tier 2* nuclear design requirements are consistent with those listed for the ABWR (incorporated by reference to Section 4B.4 of the ABWR DCD, Revision 4). The nuclear design requirements related to fuel storage and mixed-vendor fuel loading were removed. DCD Section 9.1 addresses the requirements regarding fuel storage criticality. Mixed-vendor fuel loading is not applicable to the initial core. Based on consistency with past certified designs and compliance with current regulatory criteria, the staff finds the Tier 2* nuclear design requirements (1) through (6) acceptable.

Nuclear design requirements (1) through (5) comply with GDC 11 in that the net effect of prompt inherent nuclear feedback characteristics in the core tend to compensate for rapid increases in reactivity when operating in the power range. With respect to nuclear design requirement (6), covered in RAI 4.3-10, the staff had concerns that this requirement is not specific to the CRD system and that shutdown margin requirements could be interpreted as including the standby liquid control system (SLCS). The applicant's response for RAI 4.3-10 resolved these concerns by clarifying that the SLCS is not included in the CRD requirements.

The response to RAI 4.3-10 stated that:

> "….if the selected rods were not neutronically coupled, then the worth of the hydraulic control unit (HCU) rod pair would be equal to the sum of the worth of the individual rods. The individual rod worth for each HCU pair would then be additive and one would conclude (stated in Part C of RAI 4.3-10) that SLCS is required to achieve sub-criticality with an HCU failure."

The staff agrees that HCU rods are loosely coupled, rod worth is not additive and sufficient shutdown margin exists in the event of an HCU failure without the need for SLCS. Based on the applicant's response; that sufficient shutdown margin exists in the case of an HCU failure. Based on the applicant's response, RAI 4.3-10 is resolved.

Critical Power Design Requirements

The revised critical power correlation section maintains a design change process, even describing criteria for establishing a new correlation. The design certification process requires that the NRC specifically review and approve the fuel assembly design, along with its critical power correlation, for the initial core loading (Cycle 1) in any facility that adopts the ESBWR certified design. Because the described change process implies that changes to the correlation are acceptable without NRC review, the revised Section 4B.3 was not acceptable.

Revision 0 of DCD Tier 2, Appendix 4B.5, stated, "99.9 percent of the rods in the core must be expected to avoid boiling transition for core-wide incidents of moderate frequency…." This criterion differs from GESTAR-II, which states, "Ninety nine point nine percent (99.9%) of the rods in the core must be expected to avoid boiling transition." In its response to RAI 4.2-7 regarding this apparent change in philosophy, the applicant stated that it would revise the text to be consistent. In a subsequent response to RAI 4.2-7, Supplement 1, the applicant decided to remove this text "because it is already covered in Chapter 15 of the DCD." Based on the applicant's response and the removal of the "moderate frequency" statement, RAI 4.2-7 was resolved.

4.2.3.2.2 Appendix 4C, Control Rod Licensing Acceptance Criteria

Revision 0 of DCD Tier 2, Appendix 4C, appears to be an overview of a control rod design change process. Appendix 4C to DCD Tier 2 should define the specific Tier 2 and Tier 2* control rod design requirements. The staff issued RAI 4.2-8 stating that: "Revision 0 of DCD Tier 2, Section 4C.1, states, "…designs meeting the following acceptance criteria are considered to be approved and do not require specific NRC review." This statement is inaccurate. The NRC must specifically review and approve the control rod design employed in the initial core (Cycle 1) in any facility that adopts the ESBWR certified design." The staff requested that the applicant define the specific Tier 2 and Tier 2* in the CRD requirements. In its response to RAI 4.2-8, the applicant agreed to revise the text by removing the implied change process. The staff reviewed and accepted the revised text in Appendix 4C to DCD Tier 2, Revision 3 and based on the applicant's response, RAI 4.2-8 was resolved.

Principal Fuel Channel Design and Performance Requirements

DCD Tier 2, Appendix 4C.1, states that the specific fuel channel design to be used in any facility that adopts the ESBWR certified design must comply with the following three principal fuel channel design requirements:

(1) During any design-basis events, including the mechanical loading from a safe-shutdown earthquake event combined with LOCA event, fuel channel damage should not be so severe as to prevent control rod insertion when it is required.

(2) Coolability will be maintained for all design-basis events.

(3) Channel bowing will not cause SAFDLs to be exceeded during normal operation and AOOs.

As a result of the reclassification of these principal design and performance requirements (after DCD Revision 3), the following text and RAI responses may refer to DCD Tier 1, Section 2.9, as opposed to the requirements' final location as Tier 2* criteria in Appendix 4C. In RAI 4.2-13 the staff requested that the applicant provide clarification whether the fuel design requirements are in fact Tier 1 requirements noting that the ESBWR licensing approach differs from that of the ABWR. In its response to RAI 4.2-13 the applicant stated that the ABWR DCD Tier 1 fuel channel design requirements are more appropriate than those defined for the ESBWR in DCD Tier 1, Revision 1. The Tier 1 fuel channel requirements were modified similarly to the fuel design requirements (as shown above). As part of this modification, the requirement "to ensure that channel deflection does not preclude control rod drive operation" was removed. Recent operating experience has demonstrated that channel bow may significantly impact control blade movement. Blade-to-channel clearance, blade design and materials, and burnup history affect channel deflection and its potential impact on blade movement. The staff finds the removal of this requirement acceptable because Tier 1 design requirements related to blade insertion, which capture potential effects of channel bow, remain for both the fuel design and control rod design. The DCD Revision 3 design criteria are in agreement with the regulatory requirements and are acceptable, therefore; based on the applicant's response, RAI 4.2-13 was resolved.

<u>Control Rod Design Requirements</u>

ESBWR DCD Tier 1, Revision 3, Section 2.9, provided the following four principal control rod design requirements:

(1) The control rod stresses, strains, and cumulative fatigue will be evaluated so that they do not exceed the ultimate stress or strain limit of the material, structure, or welded connection.

(2) The control rod will be evaluated to be capable of insertion into the core during all modes of plant operation within limits assumed in plant analyses.

(3) The material of the control rod will be compatible with the reactor environment.

(4) The plant core analyses will include the reactivity worth of all control rods.

DCD, Revision 3, Section 2.9, and Appendix 4C.1 to DCD Tier 2 include a control rod design requirement that states, "…lead surveillance control rods may be used." The staff issued RAI 4.2-10 requesting clarification because; whether in a Tier 1, Tier 2, or Tier 2* design requirement, the use of the term "may" needs to be revisited. In other words, there should always be an indication of this type or magnitude of design change if it would warrant in-reactor service before batch implementation. In its response to RAI 4.2-10, the applicant decided to remove any design requirements related to lead surveillance of control rods. Because of the requirement that control rod designs are NRC reviewed and approved, the staff accepted the deletion and based on the applicant's response, RAI 4.2-10 was resolved.

DCD Revision 3, Section 2.9, and Appendix 4C.1 to DCD Tier 2 define principal design criteria for the control rod. One of the design criteria (in a previous revision) stated that the stresses, strains, and cumulative fatigue will be evaluated so that they do not exceed the ultimate stress or strain limit of the material. Certain boiling-water reactor (BWR) control rod designs include long axial welds between the square tubes and welds connecting the absorber wings to the handle and connector. The staff issued RAI 4.2-14 to request that the applicant demonstrate that the structural properties (e.g. weld regions) are never more limiting than the material properties. In their response regarding the structural properties versus material properties of the control rod, the applicant agreed to revise the design requirement to include the structure and welded connection. The applicant also described mechanical testing that demonstrated that the base material failed before any of the welds. Based on the applicant's response the staff finds design requirement (1) above acceptable, therefore; RAI 4.2-14 was resolved.

DCD Revision 3, Section 2.9, in its discussion of principal design criteria, states, "The material of the control rod will be compatible with the reactor environment." In RAI 4.2-11 the staff noted in recent years the phenomena of shadow corrosion has been identified. Those phenomena are partly due to the interaction between the Zircaloy channels and stainless steel control blades. The staff requested that the applicant discuss the implementation of this design criterion with respect to shadow corrosion. In response to RAI 4.2-11 regarding the implementation of the principal design criteria with respect to shadow corrosion, the applicant stated that this design requirement was related to stress-corrosion cracking (SCC) resistance of the material and deformation induced by B_4C swelling. In its response, the applicant discussed shadow corrosion, its effect on channel bow, and the applicant's strategy for mitigating the effects of shadow corrosion. Based on the applicant's response to RAI 4.2-11 the staff finds that design requirement (3) is fulfilled, therefore; RAI 4.2-11 was resolved.

Design requirements (2) and (4), related to control rod insertion and worth, are consistent with the regulatory criteria and are acceptable.

Initially, GEH included the fuel and control rod design requirements as Tier 1; later, GEH incorporated them in the DCD as Tier 2*.

The revised Tier 2* control rod design requirements are listed below:

(1) Control rod stresses, strains, and cumulative fatigue are evaluated not to exceed the ultimate stress or strain limit of the material, structure, or the welded connection.

(2) The control rod design is evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.

(3) Control rod materials are shown to be compatible with the reactor environment.

(4) Control rod reactivity worth is included in the plant core analyses.

4.2.4 Conclusion

Based on the discussion above, the staff finds that the ESBWR DCD Tier 2 and Tier 2* criteria related to fuel system design and performance requirements (including the control assembly design) satisfy all of the regulatory requirements and SRP guidelines identified in Section 4.2.1, including the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"; GDC 10, 27, and 35; and 10 CFR 52.47(a).

As identified in Section 4.2.1 of this review, the specific fuel, control rod, and core designs referenced within the DCD will constitute an approved design that may be used for the COL first-cycle core loading without further NRC staff review. To fulfill these regulatory requirements, the ESBWR DCD references the following NRC-approved topical reports:

- NEDC-33240P and NEDC-33242P, concerning the GE14E fuel assembly design
- NEDE-33243P and NEDE-33244P, concerning the Marathon control blade design
- NEDC-33326P, concerning the GE14E initial core nuclear design

The staff has confirmed that the cited GE14E fuel assembly design and Marathon control blade design satisfy the design and performance requirements specified for the ESBWR in Appendices 4B and 4C. As such, the staff finds the use of this fuel design system acceptable for ESBWR Cycle 1. The staff's approval of the GE14E fuel assembly design and Marathon control blade design includes limitations and conditions, which are addressed in the safety evaluations for LTRs NEDC-33240P/NEDC-33242P and NEDE-33243P/NEDE-33244P.

4.3 <u>Nuclear Design</u>

4.3.1 Regulatory Criteria

The staff reviewed DCD Tier 2, Section 4.3, "Nuclear Design," in accordance with SRP Section 4.3, "Nuclear Design." DCD Tier 2, Section 4.3, presents the ESBWR nuclear design bases. The nuclear design must ensure that the SAFDLs will not be exceeded during normal operation, including AOOs, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core, or sustain unstable core conditions. To meet these objectives, the nuclear design must conform to the following GDCs:

- GDC 10, "Reactor Design," requiring the reactor design (reactor core, reactor coolant system, control and protection systems) are designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs;
- GDC 11, "Reactor Inherent Protection," requiring a net prompt inherent negative feedback power coefficient in the operating range;

- GDC 12, "Suppression of Reactor Power Oscillations," requiring that power oscillations that can result in conditions exceeding SAFDLs are not possible, or can be reliably and readily detected and suppressed;

- GDC 13, "Instrumentation and Control," requiring a control and monitoring system to monitor variables and systems to assure adequate safety including those that can affect the fission process over their anticipated ranges for normal operation, AOOs, and accident conditions;

- GDC 20, "Protection System Functions," requiring, in part, a protection system that automatically initiates a rapid control rod insertion to ensure that fuel design limits are not exceeded as a result of AOOs;

- GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," requiring protection systems designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems;

- GDC 26, "Reactivity Control System Redundancy and Capability," requiring, in part, two independent reactivity control systems of different design principles that are capable of holding the reactor subcritical under cold conditions;

- GDC 27, "Combined Reactivity Control Systems Capability," requiring, in part, a control system designed to control reactivity changes during accident conditions in conjunction with poison addition by the emergency core cooling system (ECCS); and,

- GDC 28, "Reactivity Limits," requiring, in part, that the reactivity control systems be designed to limit reactivity accidents so that the reactor coolant system boundary is not damaged beyond limited local yielding.

The acceptance criteria in the area of nuclear design, specifically power distributions, are based on meeting the relevant requirements of the GDCs related to the reactor core and the reactivity control systems.

The nuclear design basis for control requirements is that SAFDLs are met during normal operation and AOOs. Therefore, the maximum linear heat generation rate (MLHGR) and the minimum critical power ratio (MCPR) constraints shall be met during operation. The MLHGR limit and operating limit MCPR (OLMCPR) are determined such that the fuel rods do not exceed licensing limits during AOOs.

The MLHGR is the maximum local linear heat generation rate (LHGR) (more specifically, that of the fuel rod with the highest surface heat flux at any nodal plane in a fuel bundle in the core). The MLHGR operating limit is bundle-type dependent, and LTR NEDC-33242 describes the determination of this limit. The LHGR is monitored to ensure that all mechanical design requirements are met. The fuel will not be permitted to be operated at LHGR values greater than those found to be acceptable within the body of the safety analysis under normal operating conditions. Under abnormal conditions, including the maximum overpower condition, the MLHGR will not cause fuel melting or cause the strain limit to be exceeded.

The MCPR is the minimum critical power ratio of all of the fuel bundles. The critical power ratio (CPR) for any bundle is the ratio of the bundle power that would result in transition boiling to the current bundle power. Therefore, the bundle with the smallest CPR has the smallest margin to transition boiling. The CPR is a function of several parameters; the most important are bundle power, bundle flow, the local power distribution, and the details of the bundle mechanical design.

The plant OLMCPR is established by considering the limiting AOOs for each operating cycle. The OLMCPR is determined such that 99.9 percent of the rods avoid boiling transition during the limiting analyzed AOO, as discussed in LTR NEDC-33237P-A.

To meet the provisions of GDC 10, the design bases affecting power distribution of the ESBWR include the following parameters:

- Under abnormal conditions (including maximum overpower), the MLHGR will not cause the fuel to exceed mechanical design limits.

- The MCPR during normal operation will remain greater than the OLMCPR to avoid boiling transition during normal operation and AOOs.

GDC 13 provides the required criteria to evaluate core monitoring. Core monitoring is performed using in-core nuclear instrumentation, in part to ensure that the core is being operated within these limits and to ensure that automatic reactivity control systems are initiated during adverse plant transients so that SAFDLs are met.

GDC 20, 25, 26, and 27 provide the required criteria for the reactivity control system. The control rod system is designed to provide shutdown margin and reactivity control of maximum excess reactivity anticipated during cycle operation. The control rods provide reactivity changes that compensate for the reactivity effects of the fuel and water density changes accompanying power level changes over the range from full load to no load and allow for control of the power distribution within the core.

GDC 12 specifies the requirements relative to reactor stability. The staff has documented its review of the compliance of the ESBWR with the provisions of GDC 12 in Section 4D of the final safety evaluation report (FSER).

The staff separately reviewed the compliance of a proposed initial core design that was submitted in March 2009 as LTR NEDC-33326P, "GE14E for ESBWR Initial Core Nuclear Design Report," Revision 1. The staff's review of the initial core nuclear design, in accordance with the aforementioned review criteria, is documented separately in the staff safety evaluation of NEDC-33326P.

4.3.2 Summary of Technical Information

Core Description

The 4,500–megawatt-thermal ESBWR core consists of 1,132 fuel bundles and 269 control blades. Several types of fuel bundles, similar except for differences in enrichment and burnable poison content, are loaded in the reference pattern. The purpose of the bundle differences is to allow for a flatter radial power distribution across the core and provide low reactivity assemblies that are similar in their neutronic behavior to partially burnt assemblies.

Core Monitoring

The ESBWR core monitoring is accomplished with several in-core nuclear instruments that cover the expected ranges for normal operation, AOOs, and accident conditions. The neutron monitoring system comprises three separate measurement systems: the source range monitor, the local power range monitor, and the automatic fixed in-core probe. The power range neutron monitoring system (PRNM) receives signals from several local detectors. These in-core nuclear instruments include the local power range monitors (LPRMs), as well as automatic fixed in-core gamma thermometers (GTs). For low powers characteristic of the source range through a normal startup (greater than 10 percent of rated thermal power), the core neutron flux is monitored using the source range neutron monitoring system (SRNM).

The LPRMs are arranged in 64 strings, each with four detectors, and distributed throughout the core. DCD Tier 2, Figure 7.2-7, shows the locations of LPRM strings. For every four-by-four array of bundles, there are four LPRM strings (one at each corner). The LPRM strings comprise four LPRM detectors that are spaced evenly axially throughout the core. The LPRM detectors are polarized fission chambers.

Inside the LPRM instrument guide tube are seven automatic fixed in-core probes (AFIPs). The AFIP is a GT instrument that is used to periodically calibrate the LPRM signal. DCD Tier 2, Figure 7.2-8, shows the axial elevation of the AFIPs. Each LPRM instrument string contains seven AFIPs. One AFIP is at the same elevation as the midplane of each of the LPRM detectors. In between each LPRM detector, there is another AFIP. The AFIPs are evenly distributed between the uppermost and bottommost LPRMs at 381-millimeter (15-inch) intervals.

To cover the entire range of normal operation, instruments are included to measure the neutron flux and monitor the fission process in the startup range. Increased instrument sensitivity is necessary to monitor the startup process when the reactor power is very low. According to DCD Tier 2, Chapter 7, Section 7.2.2.2.4.1, the SRNM comprises 12 detectors. These detectors are fixed in-core regenerative fission chamber sensors. The 12 detectors are spaced evenly throughout the core and located at the core midplane axially; DCD Tier 2, Figure 7.2-6, shows the radial locations. The detectors are housed within pressure barrier tubes. The SRNM detectors are capable of measuring the reactor flux over ten decades, from a flux level of approximately 10^3 neutrons per square centimeter ($n/cm^2/s$) to 10^{13} $n/cm^2/s$. This range extends to approximately 10 percent of rated power. The LPRM monitoring capability overlaps this range, as the LPRMs can monitor core power from the startup range through the power range, from 1 percent of power to greater than rated thermal power.

The rod control and information system (RC&IS) is non-safety-related. The RC&IS is a logic system that provides controls on reactor maneuvering through control rod motion during normal operation and maintains status information regarding the current control rod configuration for the reactor.

Using local power indications from the LPRM detectors, the RC&IS subsystems issue rod blocks to ensure that safety and operating limits are not exceeded as a result of control rod motion. The automated thermal limit monitor and multichannel rod block monitor work together above the low power setpoint to ensure that rod withdrawals are inhibited when local detectors indicate power changes that challenge the MLHGR limit or the OLMCPR. The multichannel rod block monitor (MRBM), unlike conventional rod block monitors, uses several channels of LPRM indications throughout the core to simultaneously monitor each region of the core where control rods are being withdrawn during ganged withdrawal sequences. Below the low-power setpoint, the rod worth minimizer (RWM) is used to compare the control rod withdrawal sequence at low power to a preprogrammed control rod withdrawal pattern. In cases where the control rods are withdrawn in a different manner, the RWM enforces control rod insertions and withdrawals at low power to reduce the available reactivity worth of a control rod and thus mitigate the consequences of a control rod drop accident during low-power operation.

Upon receipt of a scram signal by the reactor protection system (RPS), the RC&IS initiates a fast fine-motion control rod drive (FMCRD) run-in as a backup to the hydraulic scram through the diverse protection system (DPS). The RC&IS also sends selected control rod run-in (SCRRI) signals to the DPS following specific AOOs, namely load rejection, turbine trip, and loss of feedwater heating.

Another important function of the RC&IS is to interface with the plant computer to perform LPRM calibration and plant simulator adaptation. This function is performed by using AFIP signals in conjunction with three-dimensional nuclear models to determine gain adjustments and nodal parameter corrections.

<u>Reactivity Coefficients</u>

The reactivity coefficients express the effects of changes in the core conditions, such as power and fuel and moderator temperature and moderator density, on core reactivity. These coefficients vary with fuel exposure and power level.

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in predicting the response of the core to external disturbances. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest are the Doppler coefficient, the void reactivity coefficient, and the moderator temperature coefficient. The combination of these reactivity coefficients dictates the power reactivity coefficient. A combination of negative coefficients ensures that the reactor will have an inherent negative reactivity feedback with increasing power.

To demonstrate that the Doppler reactivity coefficient remains negative in the power operating range, the applicant calculated temperature-dependent eigenvalues for each of the fuel bundle types. The Doppler reactivity coefficient is predominantly driven by the uranium-238 and plutonium-240 content in the fuel, and while an inherent feature of the fuel, this coefficient does not vary significantly between BWR fuel designs. The Doppler coefficient calculated for the ESBWR initial core is negative for increasing fuel temperature and similar in magnitude to operating reactor Doppler coefficients.

The void reactivity coefficient was estimated for both the power range of operation and for cold shutdown conditions. The applicant's analyses indicate a negative trend of core eigenvalue with increasing core average void content in the power range of operation, indicating inherent negative reactivity feedback under these conditions.

In RAI 4.3-6 the staff requested verification that the calculated values of the void reactivity coefficient at the beginning of cycle (BOC), middle of cycle (MOC) and end of cycle (EOC) at nominal operating conditions are negative. In response to RAI 4.3-6, the applicant provided the BOC, MOC, and EOC void reactivity coefficients predicted by PANACEA based on enthalpy perturbations to the core model. The staff finds that the results indicate a consistently large, negative void reactivity coefficient.

The magnitude of the void reactivity coefficient, however, decreases with decreasing void content. Therefore the applicant identified the cold shutdown condition as a limiting case, particularly the EOC following depletion of burnable poisons. The analysis for the limiting condition verifies that the void reactivity coefficient is negative.

Lastly, the applicant calculated the moderator temperature coefficient. During normal operation the coolant is subcooled only near the core inlet and remains at a near constant temperature once reaching saturated conditions. The EOC for the reference core loading was identified as the condition with the least negative moderator temperature coefficient. The results indicate that, at temperatures above 150 degrees C (approximately 300 degrees F), the core eigenvalue decreases with increasing water temperature.

The moderator temperature coefficient decreases in magnitude over cycle exposure with the withdrawal of control rods and the depletion of gadolinia burnable poisons. Late in the cycle, the reduction in the poison content leads to potential conditions where the reactor is overmoderated, thereby yielding a positive moderator temperature coefficient for cold conditions. While the EOC moderator temperature coefficient is positive, it is small compared to the effects of the void reactivity feedback. The applicant's calculations show that the moderator temperature coefficient at the EOC under cold conditions may be positive.

At cold conditions towards the EOC, the ESBWR neutron spectrum is slightly overmoderated, yielding a slightly positive moderator temperature coefficient for cold conditions at the EOC. In RAI 4.3-5, the staff requested additional information regarding the moderator temperature coefficient that is slightly positive at low temperatures and EOC. (The moderator temperature coefficient remains negative for all operating conditions at and above hot standby.) In their response to RAI 4.3-5 the applicant stated that the moderator temperature coefficient may become positive when the reactor coolant is below rated pressure and temperature, but during these conditions, a positive moderator temperature coefficient is manageable. Below rated conditions, the reactor power is low, and therefore, the time it takes to heat the volume of coolant to result in an appreciable increase in temperature is very long. In addition, the cooling rate for the fuel would be slow if a power increase occurred, particularly since the heatup adds negative reactivity through the Doppler Effect. Based on the applicant's responses, the staff concludes that for all operating conditions (with temperatures above hot standby) the moderator temperature reactivity coefficient is negative. At temperatures below hot standby the Doppler reactivity coefficient provides a prompt feed back to counter power increases. Therefore; based on the aforementioned responses RAIs 4.3-5 and 4.3-6 are resolved.

Reactivity Control Systems

The control rod system is designed to provide shutdown margin and reactivity control of maximum excess reactivity anticipated during cycle operation. The control rods provide reactivity changes that compensate for the reactivity effects of the fuel and water density changes accompanying power level changes over the range from full load to no load and allow for control of the power distribution within the core.

In addition to providing the means for controlling core reactivity for power maneuvering, the control rods provide the minimum shutdown margin following any AOO and are capable of making the core subcritical rapidly enough to prevent exceeding SAFDLs. The control rods are automatically hydraulically inserted upon receipt of a scram signal from the RPS.

The applicant has provided an analysis in DCD Tier 2, which shows that the control rod worth is sufficient to ensure a subcritical configuration for xenon-free, cold shutdown conditions at BOC. The BOC condition is limiting in terms of available shutdown margin.

The control rods are backed-up by the standby liquid control system (SLCS). The SLCS is a second reactivity control system meant to provide a diverse and redundant capability to the control rods. The SLCS is an accumulator-driven boron injection system. It is designed to provide the capability of bringing the reactor, at any time in a cycle, from full power with a minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state if the control rods fail to insert.

DCD Tier 2, Section 4.3, provides analyses of the shutdown capability of both the control rod system and the SLCS. The analyses show that either system is capable of holding the reactor subcritical at the limiting conditions in terms of exposure, temperature, and xenon. In the case of the control rod system, the calculations consider a single failure of a rod to insert and the single failure of a hydraulic control unit (HCU) to insert a pair of rods.

4.3.3 Staff Evaluation

Core Monitoring

The neutron monitoring system is designed to meet the requirements of GDC 13 and GDC 10. Specifically, the PRNM and SRNM are designed to monitor the fission process during normal operation and over the range of anticipated operation and accident conditions. The PRNM comprises several LPRM detectors with the capability of monitoring the neutron flux in the reactor between 1 percent of rated core power and well over 100 percent of the rated core power (125 percent). The SRNM is designed to monitor the neutron flux at very low levels (approximately 10^3 n/cm^2/s) or approximately 10 decades below the normal operating level. The combination of these two neutron monitoring subsystems allows for an overlapping monitoring capability over the full range of neutron flux levels under normal operation, including startup and AOOs. The LPRM capability extends to higher neutron flux levels, which allows for monitoring of the reactor core power during accident conditions and anticipated transients without scram. Therefore, the staff finds that the ESBWR neutron monitoring system is acceptable in that it provides sufficient capability to adequately monitor the neutron flux levels in the reactor over the necessary ranges.

The in-core ESBWR neutron monitoring system is based on a series of distributed LPRMs. The polarized fission gas chambers are substantially the same as those instruments widely applied within the operating fleet of BWRs. The design differences between the ESBWR and conventional BWRs will not impact the fundamental operation of the LPRMs so long as the steady-state bypass void fraction remains below 5 percent as described in NEDO-33239P, Revision 4 “GE14 for ESBWR Nuclear Design Report.” issued March 2009.

These instruments also interface with the 3D MONICORE system to determine the operating characteristics of the core. For the 3D MONICORE system to accurately assess the thermal margin during operations and to ensure that the RPS accurately detects adverse transient or accident conditions and initiates automatic protective actions such as scram, the instruments must be periodically calibrated.

The neutron monitoring system includes in-core GTs to replace the function of the traversing in-core probe system for conventional reactors. The GTs, much like gamma traversing in-core probe instruments, are used to determine the axial power shape and LPRM gain adjustment factors based on local gamma flux indications. The primary difference between the instruments is that the GTs are distributed, stationary probes.

The staff reviewed the information provided by the applicant concerning the GT design and found that, with regular calibration, the GT can be used to determine the local gamma flux. When combined with coupled transport calculations to determine the detector response kernels (or signal to power ratios), the GT indication may be used to adequately determine the local nodal power in surrounding nodes. NEDO-33197, Revision 2, “Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring,” issued August 2008, describes the NRC-approved methodology for translating the GT signals to power distribution information.

The GT instruments are spaced within the core beside the LPRMs, giving a complete radial mapping capability if the core power distribution is quadrant symmetric.

The 3D MONICORE system determines the margin to limits based on input from the neutron monitoring system, and input from the core thermal hydraulic instrumentation (i.e., core flow). The 3D MONICORE system is based on the PANAC11 calculational engine. NEDC-33239P-A, Revision 4, issued March 2009, describes the NRC-approved PANAC11 methodology.

However, GDC 13 also requires that appropriate controls be in place to ensure that the reactor core is operated within prescribed safety and operating limits. The GDC 13 requirements for the Neutron Monitoring System are fulfilled by prescribing limits that account for instrument and measurement uncertainties. Of key importance to the prescription of these limits is the accuracy of the neutron flux measurements. The pedigree of LPRM measurements in particular is related to the efficacy of the AFIPs and process computer to effectively and accurately calibrate the local indications of the neutron flux level. The staff issued RAI 4.2-12 and RAI 4.3-2 to request additional information regarding the determination of the MPLHGR value and the uncertainties in the nuclear instrumentation, calibration, biases, and the 3D MONICORE PANAC11 calculations. The staff reviewed the responses to RAI 4.2-12 and RAI 4.3-2 and the results of the review and the approval of these uncertainties are in the safety evaluation for NEDC-33239P-A and NEDE-33197P-A. (NEDC 33239P-A is the GE14 ESBWR Nuclear Design Topical Report and NEDC 33197P-A is the Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring) The uncertainties were correctly evaluated and properly applied to the operating limits. The in-core instrumentation meets the requirements of GDC 13 by providing monitoring capability over the range of expected operation and providing sufficient information, given the capabilities of the 3D MONICORE system, to monitor core operating parameters relative to associated operating limits. Therefore; based on the applicant's responses, RAIs 4.2-12 and 4.3-2 are resolved.

Maintaining the reactor within the OLMCPR and operating MLHGR limit ensures that the SAFDLs are not exceeded during normal operation or as a result of AOOs, therefore; the staff finds that the design basis satisfies GDC 10.

In summary, the staff finds that the ESBWR design adequately meets the requirements of GDC 10 and GDC 3 and is therefore acceptable.

Reactivity Coefficients

As described above, GDC 11 requires that the core be designed with inherent negative reactivity feedback with rising: power, fuel temperature and core void.

The applicant provided several analyses to indicate the nature and magnitude of the reactivity feedback coefficients for the reference ESBWR core. NEDC-33239P-A describes the NRC approved nuclear methods. In each case, the applicant performed the analysis by perturbing the steady-state calculation to determine the change in eigenvalue as a result of a change in the fuel temperature, coolant temperature, or coolant void.

In general, the Doppler coefficient is a strong function of fertile heavy metal content (e.g., uranium-238) and spectrum hardness. For the ESBWR, the enrichment and planar fuel geometry are similar to operating BWRs. However, the bundle pitch is slightly greater for the ESBWR compared to operating BWRs, which leads to a softer neutron spectrum arising from increased moderation in the core bypass. The softer spectrum reduces the resonance integral and, consequently, the Doppler coefficient. The applicant's calculations are consistent with this expectation.

The increased assembly spacing also affects the moderator temperature coefficient. The increased hydrogen to heavy metal ratio decreases the magnitude of the moderator temperature coefficient and leads to slightly positive values for cold conditions at the EOC where the neutron spectrum is very soft (thus, overmoderated). The positive nature of the moderator temperature coefficient is of minor concern because of the relatively slow nature of the moderator temperature change (relative to fuel temperature change), and at normal operating conditions, the core dynamic behavior is driven predominantly by the strong, negative void reactivity feedback. This condition is only for low temperatures and is not of sufficient magnitude to cause operational concerns during startup and shutdown operations, or a reactivity insertion problem.

The applicant provided a series of core calculations to determine the estimated void coefficient. As the void reactivity coefficient is stronger for higher void fractions, the applicant performed calculations for cold shutdown conditions. This calculation is conservative because the spectrum at cold shutdown conditions is overmoderated. The applicant simulated the effects of voids in the subcooled coolant using the NRC approved PANAC11 method and found that, in the most limiting case, the void reactivity coefficient was negative.

The power reactivity coefficient is a combination of the Doppler, void, and moderator temperature reactivity coefficients. While the design differences of the ESBWR make the moderator temperature and Doppler coefficients less negative than for an operating BWR, the increased void, higher enrichment, and higher burnable poison loading result in an overall negative power coefficient. In the case of the ESBWR, the void coefficient is not significantly different from operating reactors and a dominant contributor to the power coefficient. The staff finds the reactivity coefficient values to have been evaluated using NRC approved methods, to be negative, and ensure a negative power reactivity coefficient, therefore, it meets the requirements of GDC 11 and are acceptable.

Reactivity Control Systems

As described above, GDC 20, 25, 26, 27, and 28 specify the requirements for the reactivity control systems.

The reactivity control worth calculations were performed using the TGBLA06 and PANAC11 codes. The applicant calculated the shutdown margin at several exposure points during the cycle to demonstrate that BOC is the limiting condition. The analysis provided ensures that the reactor remains subcritical with sufficient margin when the strongest rod and strongest rod pair are fully withdrawn.

On this basis, the staff has determined that the control system has adequate negative reactivity worth to ensure shutdown capability, assuming that the most reactive control rod is stuck in the fully withdrawn position.

The control rod system automatically inserts control blades to shut down the reactor on receipt of a scram signal. The negative reactivity worth of the control rods is sufficient to bring the reactor to a cold-shutdown condition at any point during exposure. The core monitoring system provides operating margin to the SAFDLs. The staff finds that the ESBWR appropriately monitors the core conditions to ensure that the effects of transients do not challenge the SAFDLs and prompts automatic scram during adverse conditions; therefore, the design meets the requirements of GDC 20.

Additionally, the applicant explains that control rod assignments to particular HCUs shall maintain sufficient distance between rods such that there is essentially no neutronic coupling between the control cells and no significant impact on the shutdown margin given a failure of a single HCU. When the reactor is shut down, the core is filled with liquid water and the mean free paths for neutrons are much smaller than at power, where the presence of voids allows for increased neutron transport during slowing down. Therefore, control cell neutronic coupling is effectively limited to nearby neighboring control cells. The assignment of control rods to HCUs, such that no HCU drives two nearby control rods would preclude neutronic coupling. Without any coupling, there is no synergistic effect of a dual control rod insertion failure, which could result in local criticality. The HCU mapping provided in DCD Tier 2, Section 4.3, indicates that control rods assigned to an individual HCU are separated by several rod locations (between five and seven rod locations). As the mean free path for even higher energy neutrons at normal operating conditions ranges on the order of 15–30 centimeters, and the mean free path is greatly reduced when the core is under cold conditions with control rods inserted, the staff finds that the HCU assignments adequately preclude the possibility of synergistic reactivity effects. Therefore, local criticality based on the failure of any particular HCU is not a concern if the remainder of the control rods inserted provide sufficient negative reactivity to ensure that the reactor is shutdown and subcritical under cold conditions at its most reactive point.

The staff therefore finds that the shutdown margin is sufficiently large to provide reasonable assurance that the requirements of GDC 25 are met considering the failure of a single rod or rod pair to insert.

In DCD Tier 2, Section 4.3.1.2 and Appendix 4B state that compliance with GDC 26 is partially demonstrated by showing margin to criticality in the most reactive cold condition with the strongest rod pair withdrawn. The staff has evaluated the calculation of the shutdown margin and reactivity margin to criticality at cold conditions assuming the strongest rod pair is withdrawn. The staff finds that the shutdown margin calculations provide reasonable assurance that the control rod system is capable of holding the reactor subcritical under cold conditions; thus, the requirements of GDC 26 are met.

The SLCS meets the requirements for diverse and redundant control systems given in GDC 26 and the combined reactivity control system requirements given in GDC 27. The staff has determined that the SLCS is adequate for bringing the reactor to a cold shutdown condition at any point in exposure and therefore acts as a fully redundant diverse and adequate control system. The system is diverse in that it is a dissolved poison, passive liquid injection system, thereby satisfying GDC 26. As the SLCS is fully capable of controlling the reactivity and is an ECCS, it provides sufficient negative worth to compensate for a partial failure of the control rod system, thereby satisfying GDC 27. The analysis indicates a large reactivity margin.

Analysis of the consequences of a postulated control rod drop accident (CRDA) demonstrates compliance with GDC 28. The staff reviewed the methodology and found it to be appropriate for the design certification analysis. The staff issued RAI 4.6-23 S02 to request that the applicant must demonstrate compliance with GDC 28 regarding pressure boundary integrity and acceptable radiological consequences in case of a control rod drop accident (CRDA). In their response to RAI 4.6-23 S01, GEH stated that the most reactive rod is assumed to get separated from the drive mechanism get caught and then drop to where the rod mechanism is. This scenario literally satisfies the provisions of GDC 28. The analyses accounted for the rod reactivity, fuel burnup, and cladding Hydrogen content and calculated a conservative value of the fuel enthalpy. The results showed that the enthalpy rise is within the limits of the curves in Appendix B of Revision 3 to SRP Section 4.2. Therefore, the design meets the requirements of GDC 28, and based on the applicant's response, RAI 4.6-23 S02 was resolved.

On the basis of its review, the staff concludes that the functional design of the ESBWR reactivity control systems meets the requirements of GDC 20, 25, 26, 27, and 28 and, therefore, is acceptable.

The staff separately reviewed and verified compliance of the proposed initial core design that was submitted as an LTR (NEDC-33326P). The staff safety evaluation for NEDC-33326P separately documents the staff's review of the initial core nuclear design, in accordance with the aforementioned review criteria.

4.3.4 Conclusions

The applicant has described the computer programs and calculation techniques used to predict the nuclear characteristics of the reactor design has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the ESBWR.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, significant excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity requirements for the equilibrium cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor at any time during the cycle, with the highest worth control rod HCU stuck in the fully withdrawn position.

On the basis of its review, the staff concludes that the applicant's assessment of reactivity control requirements over the equilibrium core cycle is suitably conservative, and that the control system provides adequate negative worth to ensure shutdown capability.

The staff concludes that the nuclear design is acceptable and meets the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. This conclusion is based on the following:

- The applicant has met the requirements of GDC 11 with respect to inherent negative nuclear feedback characteristics in the power operating range by calculating a negative power coefficient of reactivity and using calculation methods that have been found acceptable.

- GDC 12 specifies the requirements related to reactor stability. FSER Section 4D documents the staff review of the compliance of the ESBWR with the provisions of GDC 12.

- The applicant has met the requirements of GDC 13 with respect to provision of instrumentation and controls to monitor variables and systems that can affect the fission process by providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns and other process variables (such as temperature and pressure), and providing suitable alarms and/or control room indications for these monitored variables.

- The applicant has met the requirements of GDC 26 with respect to provision of two independent reactivity control systems of different designs by having a system than can reliably control AOOs, having a system that can hold the core subcritical under cold conditions, and having a system that can control planned, normal power changes.

- The applicant has met the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with poison addition by the ECCS of reliably controlling reactivity changes under postulated accident conditions by: providing a movable control rod system and a liquid poison system, and performing calculations to demonstrate that the core has sufficient shutdown margin with the highest worth stuck rod.

- The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by demonstrating that the consequences of a postulated CRDA are sufficiently benign that the limits specified in SRP Section 4.2 are not challenged.

- The applicant has met the requirements of GDC 10, 20, and 25 with respect to SAFDLs by providing analyses demonstrating that normal operation, including the effects of AOOs, meet fuel design criteria, that the automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of AOOs and ensures the automatic operation of systems and components important to safety under accident conditions, and that no single malfunction of the reactivity control system causes violation of the fuel design limits.

4.4 Thermal and Hydraulic Design

In its review of the ESBWR thermal-hydraulic design, the staff considered information contained in the DCD, responses to the staff's RAIs, and the topical reports referenced by the applicant. In addition, the staff conducted its review in accordance with the guidelines provided by SRP Section 4.4, "Thermal and Hydraulic Design," Revision 2, issued March 2007.

As described in the following sections, the thermal-hydraulic design of the reactor core provides adequate heat transfer compatible with the heat generation distribution in the core.

4.4.1 Regulatory Criteria

DCD Tier 2, Section 4.4, “Thermal and Hydraulic Design,” presents the ESBWR thermal-hydraulic design bases and functional requirements of the fuel, core, and reactivity control system. The principal thermal-hydraulic design basis for the ESBWR reactor core is to ensure adequate heat removal to prevent fuel damage during any conditions of normal operation, including the effects of AOOs. GDC 10 specifies that the reactor core and associated coolant, control, and protection systems must be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. Section 4.3 and Appendix 4D to DCD Tier 2 discuss the thermal-hydraulic stability performance of the reactor. Sections 4.3, 4.A, and 21.6 of this report address the requirements of GDC 12.
Acceptance criteria are based on the following GDC:

- GDC 10, as it relates to the reactor core and associated coolant, control, and protection systems being designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and,

- GDC 12, as it relates to the reactor core and associated coolant, control, and protection systems being designed to ensure that power oscillations which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

Additionally, the staff considered the regulatory guidance in the following documents:

- SRP Section 14.3.4, “Reactor Systems—Inspections, Tests, Analyses, and Acceptance Criteria,” issued March 2007;

- Three Mile Island (TMI) Action Item II.F.2 of NUREG-0737, “Clarification of TMI Action Plan Requirements,” issued November 1980, for instrumentation provided for indication of inadequate core cooling;

- design description and proposed procedures for use of the loose parts monitoring system (LPMS), consistent with the guidance of Regulatory Guide (RG) 1.133, Revision 1, “Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors,” issued May 1981 (see Section 4.4.5 of this report); and,

- preoperational and initial startup test program recommendations of RG 1.68, Revision 3, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” issued March 2007.

The staff’s review covered the thermal-hydraulic design of the core and reactor coolant system to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, and (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs. The review also assessed the methods used to determine hydraulic loads on the core and reactor coolant system components during normal operation and design-basis accident conditions. Section 3.9.5 of this safety evaluation report discusses component structural evaluation.

SRP Section 4.4 contains the specific review criteria used by the staff in its review. The staff review included the portions of DCD Tier 1, Section 2.1.1, “Reactor Pressure Vessel and Internals” and DCD Tier 1, Section 2.1.2, “Nuclear Boiler System,” related to thermal-hydraulics. The Tier 1 design information submitted by the applicant includes the top-level design features and performance standards that pertain to the safety of the plant and include descriptive text and supporting figures. The Tier 1 information has been derived from Tier 2.

Also considered in the staff evaluation are the core safety limits and their respective bases. These appear in DCD Tier 2, Chapter 16, “Tier 2 and Generic Technical Specifications,” and Section 16B, “Generic Technical Specifications Bases.”

4.4.2 Summary of Technical Information

The ESBWR design is similar to that of the operating BWRs, except that the recirculation pumps and associated piping are eliminated. Circulation of the reactor coolant through the ESBWR core is accomplished via natural circulation. The natural circulation flow rate depends on the difference in water density between the downcomer region and the core region. The core flow varies according to the power level, as the density difference varies with changes in power levels.

To optimize flow with minimal resistance, fuel assemblies for the ESBWR design are shorter than those of operating BWRs by approximately 2 feet. Because of this, grid spacer separation and part-length rod height vary from those of conventional BWR fuel assemblies, resulting in differing flow patterns within the fuel bundles.

ESBWR DCD Tier 2, Section 4.4, describes the ESBWR design bases and functional requirements used in the thermal-hydraulic design of the fuel, core, and reactivity control system and relate these design bases to the applicable GDC. Thermal-hydraulic design of the core shall establish the thermal-hydraulic safety limits for use in evaluating the safety margin in accordance with GDC 10. The objective for normal operation and AOOs is to maintain nucleate boiling and thus avoid a transition to film boiling. Limits are specified to maintain adequate margin to the onset of the boiling transition. The key parameter used for plant operation is the CPR, or the ratio of the bundle power at which some point within the assembly experiences onset of boiling transition to the operating bundle power. Thermal margin is stated in terms of the MCPR that corresponds to the most limiting fuel assembly in the core.

DCD Tier 2, Section 4.4, references NEDC-33237P, “GE14 for ESBWR—Critical Power Correlation, Uncertainty, and OLMCPR Development,” for discussion of the development and application of the General Electric Critical Quality Boiling Length (GEXL)14 critical power correlation for the ESBWR GE14E fuel. The GEXL14 correlation has been used for evaluation of the commercially available GE14 fuel, a conventional 12-foot long, 10x10 fuel bundle design. The shortened ESBWR (GE14E) fuel assemblies will use components identical to those used in the GE14 fuel design. These include lower and upper tie plates, grid spacers, and water rods. As described in NEDC-33237P, the

GEXL14 correlation was originally developed using full-scale test data obtained from the ATLAS critical power test facility. This facility used an electrically heated mockup of a BWR fuel bundle containing prototypical spacers and operating at conventional BWR flow rates, pressures, and temperatures. The NRC staff approved the use of the GEXL14 correlation for conventional GE14 fuel in the FSER issued August 3, 2007, for GNF topical report NEDC-32851P, Revision 2, "GEXL14 Correlation for GE14 Fuel," issued September 2001. Similar critical power tests have been conducted at the Stern Laboratories test facility in Hamilton, Ontario, using a full-scale mockup of a GE14E bundle, with operating conditions expected for the ESBWR. NEDC-33413P, "Full Scale Critical Power Testing of GE14E and Validation of GEXL14," issued March 2008, documents these tests and their statistical evaluation.

The critical power and pressure drop tests conducted for a simulated GE14E fuel bundle validate the use of the GEXL14 correlation and demonstrate the adequacy of the established GEXL14 statistics for the GE14E fuel. NEDC-33413P provides the details of the test facility, test matrix, test results, and GEXL14 statistical analysis.

To evaluate the effect of design differences between the GE14E and GE14 fuel, the applicant has used the steady-state sub-channel analysis computer code COBRAG. COBRAG is used to predict bundle critical powers and dryout locations, bundle averaged and planar local void fractions, and bundle pressure drops in BWR fuel bundles. GEH submitted the COBRAG code to the NRC staff to enable the staff to assess the sensitivity of the GE14E fuel design to spacer locations and part-length rod height within the fuel bundle. The NRC staff validated the applicability of the code independently by comparing the code's predictions to benchmark fuel data. The validation results confirmed GEH's claim that the COBRAG code is an appropriate computational tool as applied to the adjustment of the GEXL14 correlation additive constants, which are used to account for variation in power between fuel rods.

Topical report NEDC-33237P provides a detailed description of the studies performed for the assessment of differences in total heated length of the fuel assemblies, grid spacer separation, and part-length rod height and presents a statistical determination of the critical power correlation uncertainties. The overall correlation uncertainty, which includes both measurement and calculation uncertainties, will be applied to all ESBWR applications where the correlation is used.

In DCD Tier 2, Section 4.4, the applicant described how the ESBWR meets GDC 10 and GDC 12 and other acceptance criteria of SRP Section 4.4 by direct reference to the fuel design acceptance criteria provided in DCD Tier 2, Appendix 4B, and NEDC-33237P. These references provide thermal-hydraulic parameters and limits related to neutronic and thermal-hydraulic aspects of the fuel design. Section 4.2 of this report documents the staff evaluation of DCD Tier 2, Appendix 4B.

A brief summary of technical information is provided below by subject.

4.4.2.1 Critical Power

The thermal-hydraulic design of the core establishes the thermal-hydraulic safety limits for use in evaluating the safety margin in accordance with GDC 10. The margin to SAFDL is maintained during normal operation when the operating limit minimum critical power ratio (OLMCPR) is greater than the safety limit minimum critical power ratio

(SLMCPR) and the LHGR is maintained below the MLHGR limit(s). The MCPR is the minimum CPR of all of the fuel bundles. The CPR for any bundle is the ratio of the bundle power that would result in transition boiling to the current bundle power. Therefore, the bundle with the smallest CPR has the smallest margin to transition boiling. The CPR is a function of several parameters; the most important are bundle power, bundle flow, the local power distribution, and the details of the bundle mechanical design.

Section 5.13 of topical report NEDC-33237P discusses this in further detail. The limits are determined by analysis of the most severe AOOs and, considering uncertainties, provide reasonable assurance that no significant fuel damage results. Thermal margin is stated as the minimum value of the CPR that corresponds to the limiting fuel assembly in the core. The design requirement is based on a statistical analysis that demonstrates that, for AOOs, at least 99.9 percent of the fuel rods would be expected to avoid reaching boiling transition.

NEDC-33237P presents the results for the bundle critical power performance. Full-scale GE14 fuel test data are used to support the development of a critical power correlation for the ESBWR fuel, GE14E. An analytical approach is provided to adjust the GE14 test data to account for the GE14E design differences, including the overall heated length of the fuel assembly, the part-length rod height differences, and the grid spacer separation differences. This approach is demonstrated to be conservative by the confirmatory critical power tests conducted on a simulated GE14E fuel bundle.

4.4.2.2 Fuel Cladding Integrity Safety Limit

The fuel cladding integrity safety limit (FCISL) is specified such that no significant fuel damage is calculated to occur during normal operation and AOOs. Although it is recognized that the onset of boiling transition would not result in damage to BWR fuel rods, a calculated fraction of rods expected to avoid boiling transition has been adopted as a safety limit. The FCISL is defined as the fraction (percent) of total fuel rods that are expected to avoid boiling transition during normal operation and AOOs. A value of 99.9 percent provides assurance that SAFDLs are met. NEDC-33237P provides the FCISL results, along with an evaluation of the uncertainties applicable to the ESBWR design. The statistical analysis model used produces a CPR map of the core, based on steady-state uncertainties that are coupled to the TRACG ΔCPR/initial critical power ratio (ICPR) results to develop the OLMCPR. Section 5.13 of NEDC-33237P gives details of the calculation procedure.

4.4.2.3 Operating Limit Minimum Critical Power Ratio

A plant-unique OLMCPR is established to provide adequate assurance that the FCISL for that plant is not exceeded during normal operation and any AOO. By operating with the MCPR at or above the OLMCPR, the FCISL for that plant is not exceeded during normal operation and AOOs. Section 5.13 of NEDC-33237P presents a detailed discussion of the OLMCPR calculation methodology.

In addition, NEDC-33237P evaluates the fuel bundle critical power performance. This report uses full-scale test data for a conventional BWR GE14 fuel bundle to justify that the same critical heat flux (CHF) correlation can be applied conservatively to the GE14E ESBWR fuel.

LTR NEDC-33413P describes the tests conducted on a simulated GE14E fuel bundle to confirm that the GEXL14 correlation, with adjustments to the additive constants to account for differences between GE14 and GE14E fuel, can be used for CPR determination of the ESBWR fuel. The tests demonstrate that the correlation is conservative when applied to GE14E fuel over the expected ESBWR operating range.

4.4.2.4 Void Fraction

The void fraction in a BWR fuel bundle has a strong effect on the nuclear flux and power distribution. The ESBWR design calculations use an empirical correlation based on the characteristic dimensions of the fuel bundle and the hydraulic properties of the flow in the bundle. The 3D core simulator code (PANAC) and the steady-state thermal-hydraulic calculations utilize the General Electric-Hitachi void correlation. Section 21.6 of this report discusses the staff evaluation of the TRACG program for ESBWR transients. LTR NEDC-33239P, "GE14 for ESBWR Nuclear Design Report," discusses the three-dimensional, quasi-steady-state core simulator model, PANAC11. The staff evaluation of the core simulator code is presented in the SER for GEH LTRs NEDC-33239P and NEDC-33197P.

The Findlay-Dix correlation (found in the General Electric Proprietary Report NEDE-21565, "New BWR Void Fraction Correlation," issued January 1977) is used in the three-dimensional core simulator and in steady-state thermal-hydraulic calculations. This approach is also described in NEDC-32084P-A, Revision 2, "TASC-03A, A Computer Program for Transient Analysis of a Single Channel," issued July 2002.

The TRACG computer program, used for transient analyses of LOCAs, anticipated transients without scrams (ATWSs), and AOOs, employs a drift flux, interfacial shear model, which is described in NEDE-32176P, Revision 3, "TRACG Model Description," issued April 2006. NEDE-32177P, Revision 2, "TRACG Qualification," issued January 2000, discusses the qualification of the TRACG program. The SER for LTR NEDE-33083P describes the staff's review and approval as it relates to ESBWR transient analysis.

4.4.2.5 Core Pressure Drop and Hydraulic Loads

The TRACG program has been used to calculate the reactor internal pressure drop and hydraulic loads during normal operation and all AOOs, infrequent events, and accidents (e.g., LOCAs). The total pressure drop consists of friction, local, elevation, and acceleration terms. The TRACG model of the reactor vessel internals consists of radial and axial nodes that represent the boundaries of internal components. They are connected by flow paths with appropriate resistance and inertial characteristics. TRACG solves the equations of conservation of mass and energy for each node, along with the momentum equation, to give depressurization rates and local pressures. Internal component loads are then calculated from the pressure differences. Approved LTR NEDE-33083P-A, Revision 0, "TRACG Application for ESBWR," issued March 2005, discusses the TRACG program flow and pressure drop models for the ESBWR design in detail. NEDE-32176P provides the theoretical development and model description. NEDE-32177P describes the TRACG program qualification.

The friction pressure drop component is calculated as a conventional two-phase pressure drop, with a single-phase friction factor and a two-phase friction multiplier. Full-scale rod bundle pressure drop data from LTR NEDC-33238P, Revision 0, "GE14 Pressure Drop Characteristics," issued December 2005, were used to validate these friction factors for GE14 fuel components, including upper and lower tie plates, grid spacers, water rods, and part-length rods. The local pressure drop component is defined as the irreversible pressure loss associated with an area change, such as an orifice, lower tie plate, and grid spacers. It is calculated in a manner similar to the friction pressure drop, except that the local loss coefficient, K, replaces the friction coefficient. The coefficients are determined by tests, as documented in NEDC-33238P.

Additional pressure drop tests were performed using a mockup of the GE14E fuel bundle, with operating conditions expected for the ESBWR. These tests were documented in LTR NEDC-33456P, "Full-Scale Pressure Drop Testing for a Simulated GE14E Fuel Bundle," issued March 2009. The ESBWR fuel bundle-specific (GE14E) critical power and pressure drop testing was performed to better characterize the thermal-hydraulic performance of the GE14E fuel, which is shorter than the conventional GE14 fuel used in currently operating BWRs. The differences include the active fuel length, the number and axial location of the fuel rod spacers, and the axial length of the part-length rods. Also, there are differences in nominal operating conditions. The nominal bundle power and flow for the ESBWR are lower than those for the current operating fleet BWRs. The spacer loss coefficients are determined for the GE14E fuel in the ESBWR application from the pressure drop test data.

The elevation pressure drop component is determined by a conventional approach, accounting for the density change over a given height. The equation appears in DCD Tier 2, Section 4.4.2.3.3. The density term is the average mixture density, with liquid and vapor components of the two-phase fluid, weighted by the void fraction, which is determined by the drift-flux model incorporated in TRACG.

The acceleration pressure drop component is a reversible pressure change that occurs when an area change is encountered, and it is an irreversible loss when the fluid is accelerated through the boiling process. DCD Tier 2, Section 4.4.2.3.4, presents the equations used in TRACG.

NEDE-33083P-A discusses detailed core pressure drop methodology for the ESBWR. Table 4.4-1a and 4.4-1b of DCD Tier 2 provide thermal-hydraulic design characteristics of the ESBWR reactor core and compare these to typical BWR/6 and ABWR values.

4.4.2.6 Core Coolant Flow Distribution

Based on the prediction of core pressure drop, the distribution of flow into the fuel channels and the core bypass regions is calculated using the TRACG program. The core coolant flow distribution forms the basis of the prediction of steady-state and transient critical power and void fraction. TRACG treats all fuel channels as one-dimensional (axial), while the vessel is modeled as a three-dimensional component. The bundle pressure drop evaluation includes frictional, local, elevation, and acceleration losses. The pressure drop methodology has been qualified to test data in NEDE-32177P. NEDE-33083P-A discusses the TRACG program flow and pressure drop models for the ESBWR design in detail. TMI Action Item II.F.2 of NUREG-0737 requires instrumentation, such as level sensors, for the indication of inadequate core cooling. DCD Tier 2, Table 1A-1, discusses the proposed RPV level instrumentation. DCD Tier 2, Chapter 7, discusses the instrumentation and control systems in detail. The ESBWR design provides for the detection of conditions indicative of inadequate core cooling by a direct water-level instrumentation system. Both wide-range and fuel zone instruments measure the coolant level in the RPV. The RPV water level is the primary variable indicating the availability of adequate core cooling. Adequate redundancy is provided from the bottom of the core support plate to the centerline of the main steamlines.

The ESBWR is designed in accordance with the most recent Revision 4 of RG 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” issued June 2006.

4.4.2.7 Fuel Heat Transfer

The heat transfer model must accurately predict heat transfer between the coolant, fuel rod surface, cladding, gap, and fuel pellet in the evaluation of core and fuel safety criteria. Conventional methods and assumptions are employed in the TRACG program, which is used for transient analyses of LOCAs, ATWSs, and AOOs. NEDE-32176P, NEDE-32177P, and NEDE-33083P-A discuss the TRACG heat transfer models in detail.

The TRACG program includes standard heat transfer regimes (single-phase liquid or vapor), nucleate boiling, CHF, transition boiling, film boiling, and condensation with and without the effect of non-condensables. The program provides correlations for transition between different heat transfer regimes. The correlations for different regimes are standard, well-accepted correlations from the literature. However, for CHF, TRACG uses the proprietary GEXL correlation (GEXL), based on the critical quality concept for normal flows. The NRC has approved the GEXL correlation for specific fuel designs, including GE14, which provides the basis for the GE14E design in NEDC-33240P, “GE14E Fuel Assembly Mechanical Design Report” issued August 2009.

During normal operation and AOOs, convection and nucleate boiling are the most significant heat transfer mechanisms between the coolant and fuel rod surfaces. The applicant used the Dittus-Boelter correlation for the single-phase convective heat transfer for both fuel design (in the core simulator code, PANAC) and systems analyses (in the TRACG code). For nucleate boiling, the applicant used the Jens-Lottes correlation for fuel design (in the core simulator code) and the Chen correlation for systems analyses (performed using TRACG). These three correlations are widely accepted in the nuclear industry for rod bundle heat transfer. The fuel rod thermal-mechanical design analysis program GSTRM incorporates the same heat transfer models, as discussed in DCD Tier 2, Section 4.2. NEDC-33239P discusses the core simulator code, PANAC.

4.4.2.8 Maximum Linear Heat Generation Rate

The adequacy of MLHGR limits is evaluated for the most severe AOOs to provide reasonable assurance that no fuel damage results during AOOs. Margin to design limits for circumferential cladding strain and centerline fuel temperature is evaluated for AOOs. Additional discussion appears in the SER for DCD Tier 2, Section 4.3. DCD Tier 2, Section 15.2, provides the AOO results.

4.4.2.9 Core Power Operating Map

DCD Tier 2, Section 4.4.4.3, states that the core power-flow map is a single line, and there is no active control of the core flow at a given power level. The applicant provided a core power-feedwater temperature operating map in DCD Tier 2, Figure 4.4-1, which increases operating flexibility. DCD Tier 2, Sections 10.4.7.2.2.3 and 7.7.3, respectively, discuss the system hardware and control system required to develop and implement such an operating domain. LTR NEDO-33338, "ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis," issued May 2009, presents a detailed discussion and analysis. FSER Section 15.1 discusses the staff review of NEDO-33338.

4.4.2.10 Inadequate Core Cooling Monitoring System

The ESBWR inadequate core cooling (ICC) monitoring system is discussed in DCD Tier 2, Appendix 1A, "Response to TMI Related Matters." TMI Item II.F.2 in Table 1A-1 (TMI Action Plan Items) addresses this issue as it relates to the ESBWR. The ESBWR ICC monitoring system provides direct water-level instrumentation, using both wide-range and fuel zone instruments. The four divisions of wide-range instruments cover the range from above the core to the main steam lines. The four channels of fuel zone instruments cover the range from below the core to the top of the steam separator.

4.4.2.11 Loose Parts Monitoring System

The applicant has withdrawn the LPMS from the ESBWR design certification for the reasons given in its response to RAIs 4.4-7 through 4.4-9. FSER Section 4.4.3 discusses the staff evaluation. Based on the applicant's responses, RAIs 4.4-7, 4.4-8, and 4.4-9 were resolved.

4.4.2.12 Testing and Verification

Chapter 14 discusses the testing and verification techniques to be used to ensure that the planned thermal and hydraulic design characteristics of the core have been provided and will remain within required limits throughout the core lifetime.

4.4.3 Staff Evaluation

The following presents the staff evaluation of core thermal-hydraulic topics discussed in DCD Tier 2, Section 4.4.

4.4.3.1 Critical Power

DCD Tier 2, Section 4.4, references LTR NEDC-33237P to justify the use of the GEXL14 correlation for ESBWR fuel (GE14E) applications and to describe the determination of the overall correlation uncertainty. This document describes the application of the GEXL14 critical power correlation to ESBWR fuel (GE14E) and the supporting analyses performed to quantify and subsequently account for the effect (on critical power) of the differences between GE14 for the conventional BWRs and GE14E for the ESBWR. The GEXL14 critical power correlation for conventional GE14 10x10 fuels was developed using data obtained from the ATLAS critical power test facility. GE14 fuel is currently being used in operating BWRs. A significant and successful operating experience base has been developed for BWRs using GE14 fuel operating at the original design rated power and those operating at extended power uprate. Because of the similarity between the conventional BWR and ESBWR versions of GE14, the applicant proposed to use the GEXL14 correlation for ESBWR applications, with adjustment for the geometry differences between the two versions of GE14.

First, the ATLAS critical power data for the conventional BWR version of GE14 is adjusted because of the shortening of the heated length of the fuel assembly. A COBRAG subchannel computer program analysis model of GE14, previously qualified based on the ATLAS GE14 critical power data, is then used to quantify the effect of the geometry differences between the two GE14 versions on the critical power performance of the ESBWR version of GE14.

The staff's review of NEDC-33237P, which includes an assessment of the critical power evaluation method described by the applicant in DCD Tier 2, Section 4.4, appears in detail in the safety evaluations for NEDC-33237P and NEDE-33413P. NEDC-33237P, Revision 4, incorporates RAI responses based on previous revisions and references test report NEDC-33413P to confirm the applicability of the GEXL14 correlation to GE14E fuel.

The staff performed confirmatory analysis of the COBRAG code studies used by the applicant to adjust the GEXL14 correlation predictions to account for the differences in grid spacer separation and part-length rod height between the GE14E fuel used for the ESBWR and the GE14 fuel used in operating BWRs. In RAI 4.4-25, the staff requested the applicant to provide the COBRAG program and input decks so that sensitivity studies could be performed with the closer grid spacer separation and the shorter part-length rod height of the ESBWR. The studies confirmed the applicant's statement that the effects on critical power of grid spacer separation and shorter part-length rod height offset each other. Based on the applicant's response and the confirmation of spacer separation and shorter part length rods relation, RAI 4.4-25 was resolved.

In RAI 4.4-1, the NRC staff requested that GEH provide detailed information regarding the following:

- analyses and testing performed to demonstrate compliance of the ESBWR with regulations;

- the means by which the design addresses the regulatory guidance outlined in SRP Section 4.4;

- justification of the applicability of traditional computational methods (if used) to the ESBWR; and,

- justification of the applicability of new computational methods (if used) to the ESBWR, as well as differences between new methods and traditional methods.

In part (a) of RAI 4.4-1, the staff requested if any analyses or tests are necessary to demonstrate compliance with the regulations and that GEH discuss the theoretical or experimental basis, the method used, the assumptions and boundary conditions, the limitations, and the results as applied to the ESBWR design. The staff agreed that the critical power performance of the GE14E fuel will be similar to that of the GE14 fuel already used in operating BWRs, since the bundle components are similar. However, the staff believed that the thermal-hydraulic response characteristics of the shorter overall length of the GE14E assemblies, and, in particular, the shorter part-length rod height, may contribute to variation in the critical power correlation uncertainties which cannot be accurately determined by computer code (COBRAG) assessment alone. Therefore, the staff requested that the applicant submit a proposed CHF (critical power performance) test matrix for the GE14E fuel and submit proposed ITAAC to ensure that CHF testing is conducted to validate the use of the GEXL14 correlation for ESBWR application before initial core loading.

In response (to part (a) of RAI 4.4-1) the applicant stated that no new testing is necessary to demonstrate compliance of the ESBWR core thermal and hydraulic design with regulations. The applicant further stated that the analysis methodologies are applicable to the ESBWR, as discussed in the revised LTR NEDC-33237P. The applicant discussed how the ESBWR design meets the regulatory guidance in Section 4.4 of the SRP by complying with GDC 10 and GDC 12. GDC 10 requires that the reactor core be designed such that fuel design limits will not be exceeded. The ESBWR conforms to GDC 10 by establishing a MCPR such that at least 99.9 percent of the fuel rods in the core would avoid boiling transition during normal operation or AOOs. Section 4.4.1.1.1 of the ESBWR DCD Tier 2 discusses conformance with GDC 10. GDC 12 requires that when fuel design limits are exceeded, power oscillations either cannot occur or can be reliably and readily detected and suppressed. The applicant stated that the stability evaluation in Appendix 4D to DCD Tier 2 sufficiently addresses GDC 12. The ESBWR is designed to maintain stability during normal operation, as well as during AOOs. As a backup, the ESBWR maintains the ability to detect and suppress instability. The ESBWR complies with GDC 12 by implementing design criteria for the decay ratio in the form of a stability map.

Additionally, the applicant stated that the TRACG code is used for the systems analysis of the ESBWR and that a core simulator code is used for the ESBWR core design. It further stated that the responses to several of the RAIs concerning Section 4.4 show that the models in both codes apply. In addition, the NRC has already approved the applicability of TRACG to the ESBWR for LOCA and stability analyses. The staff documented its review of TRACG application for an ESBWR LOCA in the "Addendum to the Safety Evaluation Report for NEDC-33083P, 'Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design,'" and the "Addendum to the Safety Evaluation Report for NEDC-33083P, Supplement 1, and 'TRACG Application for ESBWR Stability.'" The staff addressed the applicability of TRACG to ESBWR ATWS and AOO analyses in the "Safety Evaluation Report for Application of the TRACG Computer Code to ATWS for the ESBWR Design, NEDE-33083P, Supplement 2" and in the "Safety Evaluation Report for Application of the TRACG Code to AOOs and Infrequent Events, NEDE-33083P, Supplement 3." Also, the applicant stated that no new computation methods are used for the ESBWR.

The NRC staff found the response to RAI 4.4-1 acceptable, with the exception of the GEH position that no new testing was necessary for the GE14E fuel because of the difference in length between the GE14 fuel assemblies (for which testing was performed) and the GE14E fuel assemblies. This exception resulted in RAI 4.4-1 S01. This supplemental RAI requested that the applicant submit a proposed CHF test matrix for the GE14E fuel and the corresponding proposed ITAAC to ensure that CHF testing is conducted before initial fuel load to validate the use of the GEXL14 correlation.

The applicant's response to RAI 4.4-1, S01, referred to full-scale GE14E testing that has been performed and provided NEDC-33413P. At the time the response was transmitted, the results of the GE14E testing were still being analyzed. Additionally, the applicant responded that "As testing has been performed sufficient to confirm the adequacy of GEXL14, it is not necessary to construct an ITAAC."

The NRC staff's review of the response to RAI 4.4-1, S01, led to RAI 4.4-1 S02 where the staff requested further justification for the following four items:

- Explain the axial power distributions used for the tests.
- Provide justification for testing only at 1,000 pounds-force per square inch absolute (psia).
- Provide the statistical assessment of the GEXL correlation uncertainty for GE14E fuel.
- Use the GE14E test data to show that the R-factor calculation methodology is applicable to the ESBWR.

The applicant's response to RAI 4.4-1 S02, discussed each of the above items.

The NRC staff review of the responses to these RAIs concluded that the applicant has sufficiently addressed RAI 4.4-1 and Supplements 1 and 2 to this RAI by the incorporation of revisions to NEDC-33237P and NEDC-33413P. Therefore, based on the applicant's responses, RAIs 4.4-1 and its Supplements 1 and 2 were resolved.

In RAI 4.4-5, the staff requested that the applicant describe the applicability of the bundle critical power performance method to the ESBWR design. DCD Tier 2, Section 4.4.2.1.1, refers to topical report NEDO-10958-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application," issued

January 1977. The staff requested that the conditions and limitations applicable to its use for the ESBWR design be specified. In response, the applicant indicated that NEDC-33237P discusses the applicability of GEXL to GE14E fuel, and a revision to that report will provide additional information. The applicant revised NEDC-33237P to include reference to the GE14E bundle-specific test data and analyses documented in NEDC-33413P. This report confirms that the critical power performance method proposed by the applicant is conservative therefore, based on the applicant's response, RAI 4.4-5 was resolved.

In RAI 4.4-19, the staff requested a description of the uncertainties referred to in the DCD and a comparison to conventional (operating) BWR uncertainties.
These include:

- uncertainty ranges of manufacturing tolerances,
- uncertainties in measurement of core operating parameters,
- calculation uncertainties,
  uncertainty in the calculation of the transient ΔCPR/ICPR and
- statistical uncertainty associated with the critical power correlations.

In response to RAI 4.4-19, the applicant updated Section 5.0 of NEDC-33237P to include a detailed discussion of the uncertainties that contribute to the overall uncertainty in the GEXL14 correlation when applied to the ESBWR. Most of the uncertainties, including pressure, flow, and temperature measurement, are not unique to the ESBWR. The staff reviewed the information provided by the applicant and determined it to be acceptable In that they are not unique to the ESBWR therefore, based on the applicant's response, RAI 4.4-19 was resolved.

RAI 4.4-26 summarizes the applicant's commitments from the closed proprietary meeting during the week of June 19, 2006, which include a revision to topical report NEDC-33237P to provide supporting test data and additional discussion of uncertainties. Specifically, the staff requested the ATLAS test data for GE14 fuel that were used in support of the adjustment of the GEXL14 correlation to account for GE14E fuel design differences, along with a more detailed discussion of the uncertainties unique to the ESBWR design and a statistical analysis using a 95/95 confidence level methodology.

The applicant responded to RAI 4.4-26 by providing a draft revision to topical report NEDC-33237P, which added Appendices A, B, and C. The response provided ATLAS critical power test data for the GE14 fuel that was used for the COBRAG studies and statistical evaluation, presented in tabular form and adjusted for the truncated length of the GE14E fuel used for the ESBWR. The applicant subsequently incorporated the draft appendices in Revision 1 of NEDC-33237P. Appendix A describes the COBRAG sub-channel analysis. Appendix B provides the ATLAS data for various pressure, mass flux, and inlet sub-cooling conditions, along with the corresponding adjusted critical power. The same table compares the COBRAG results considering grid spacer separation differences and part-length rod height differences of the GE14E fuel. The GEXL14 correlation prediction for the same test conditions appears in a separate column.

In RAI 4.4-26, the staff requested the applicant to explain the discrepancy in Table A-1 of NEDC-33237P, the GEXL14 10x10 COBRAG/ATLAS critical power category, and the supporting database provided in Table B-1. In addition, the staff requested the applicant to provide information for the individual assemblies missing from Table A-2 and, for Table A-5, to state what database was used to derive the ATLAS cosine standard deviation. The staff noted that the applicant should identify the test runs for Table C-1, and in Section A.2, the applicant should explain which data are used and how they are applied. The applicant provided a revision to NEDC-33237P which incorporated the corrections and the additional information. The staff reviewed the revised topical report and conducted its own internal calculations to confirm the applicant's results. The staff concluded that the applicant's statistical analysis regarding the ESBWR critical power correlation and the associated uncertainties is acceptable because they are comparable with those estimated by the staff therefore, based on the applicant's response, RAI 4.4-26 was resolved.

4.4.3.2 Fuel Cladding Integrity Safety Limit Minimum Critical Power

In conventional BWRs, Technical Specification (TS) 2.1.1 specifies the FCISL as an SLMCPR. The FCISL depends on the operating mode, the reactor steam dome pressure, and the core flow as a percentage of rated core flow. TS 2.1.1 provide a MCPR for allowed operation. If the condition is not met, the reactor must be shut down. The ESBWR TSs are based on the standard BWR/6 TSs in NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Revision 3, Volume 1. DCD Tier 2, Chapter 16, and the corresponding bases in Chapter 16B provide reactor core safety limits in TSs 2.1.1 and B2.1.1. The FCISL is designed such as 99.9 percent of the fuel rods expected to avoid boiling transition. This differs from the SLMCPR specification for conventional BWRs. In response to RAI 15.0-16, the applicant provided justification for using the number of rods subject to boiling transition as a safety limit to replace the SLMCPR in the ESBWR TS.

The staff disagreed with the applicant's position. The staff concluded that the SLMCPR numerical value should be kept as a safety limit in the TS as in the BWR standard TS. In RAI 15.0-16 S01, the staff requested that the original TS safety limit be restored. In a subsequent DCD revision, the applicant restored the SLMCPR value. FSER Section 15.1.1 presents the detailed staff evaluation. Based on the applicant's response that restored the original TSs, RAI 15.0-16 was resolved.

DCD Tier 2, Section 4.4.3.1.2, refers to Section 6 of LTR NEDC-33237P for a summary of the basis for the representative OLMCPR used for the ESBWR to protect the FCISL. Section 5 of the topical report describes the basis for the uncertainties specific to the ESBWR used in this evaluation.

The staff has evaluated the applicant's methodology for the determination of the OLMCPR and FCISL and has performed a confirmatory evaluation of the applicant's COBRAG studies to investigate the effects on critical power due to closer grid spacer separation and shorter part-length rod height of the ESBWR GE14E fuel. The staff finds the applicant's general approach acceptable. The staff's SER for NEDC-33237P include staff evaluation of this issue.

4.4.3.3 Operating Limit Minimum Critical Power Ratio (OLMCPR)

The staff performed a confirmatory evaluation of the COBRAG grid spacer separation and part-length rod height studies used by the applicant to adjust the approved conventional GEXL14 correlation for the GE14E fuel differences. As noted above, the staff finds the applicant's approach acceptable.

The staff performed confirmatory studies to evaluate the use of the GEXL14 correlation for the GE14E fuel of the ESBWR. In RAI 4.4-25, the staff requested the applicant to provide the COBRAG program executable and identify the computer platform and operating system, the user manual, and the code input decks used for the geometry difference studies (including spacer separation and part-length rod height). In response to the RAI, the applicant provided the staff with the COBRAG computer program, operating system, and user manual, therefore, RAI 4.4-25 is closed.

To assess GEH's application of the COBRAG code for the determination of the effects of closer spacer separation and shorter part-length rods on the ESBWR GE14E fuel, the NRC staff ran a number of parametric cases that approximated the sensitivity studies performed and documented by the applicant in LTR NEDC-33237P. This topical report outlines a procedure by which the ATLAS critical power data collected for the 12-foot GE14 fuel was adjusted using COBRAG to account for geometry differences of the GE14E fuel. This GE14E fuel assembly model was then used to determine the separate and combined effects on critical power of the GE14E spacer locations and part-length fuel rod length.

To approximate the parametric studies presented by the applicant, the staff constructed COBRAG input decks that varied the relevant parameters, starting with the input deck provided with GE14E.

When completed, the modified input decks were executed using GEH's version of COBRAG. The resulting output was compared to the results of Run No. 156 in Enclosure 1 of the RAI 4.4-25 response, and differences between the two datasets were found to be less than 1 percent. These small discrepancies were expected and are products of slight modeling differences between the NRC staff's approach and that of the applicant. Nevertheless, the overall trends as presented in the topical report in general, and for Run No. 156 in particular, are in agreement with the staff's results. Differences between the staff's and the applicant's predictions were deemed negligible, consistent, and due to modeling differences, therefore, based on the applicant's response, RAI 4.4-25 was resolved.

The staff's confirmatory studies do not constitute a formal staff review of the COBRAG program. COBRAG was submitted on the ESBWR docket by Global Nuclear Fuels letter FLN-2007-023, dated July 5, 2007. It was later withdrawn (MFN 08-218), since the GEXL14 correlation additive constants applicable to the ESBWR GE14E fuel can be derived from the critical power test measurements for simulated GE14E fuel.

In RAI 4.4-27, the staff requested a discussion of the applicability of an approved topical report, NEDC-32505P-A, Revision 1, "R-Factor Calculation Method for GE11, GE12, and GE13 Fuel," issued July 1999, used for R-factor determination for conventional BWR GE11, GE12, and GE13 fuel bundles, to the ESBWR GE14E fuel bundle design. The staff agreed that the same methodology can be used to calculate the GE14E bundle R-factor, but the applicant should provide confirmation. The applicant responded that no new features affecting the R-factor methodology were introduced with the GE14E fuel design. In response to RAI 4.4-27 S01, the applicant provided additional qualitative and quantitative discussion, as requested by the staff. The NRC staff accepted the supplemental response; with the condition that the R-factor will be reassessed (RAI 4.4-27 S02) and the methodology satisfactorily confirmed when the new critical power performance data are collected for the ESBWR GE14E fuel assembly ITAAC. NEDC-33413P, submitted in response to RAI 4.4-27 S02, summarizes the test results for GE14E bundle-specific critical power tests and provides confirmation that the R-factor methodology used is acceptable. Additionally, the tests confirm that the method proposed by the applicant to evaluate the critical power performance of GE14E fuel is conservative. Since the GE14E fuel bundle-specific tests have been successfully completed, there is no longer a need to establish fuel critical power performance ITAAC therefore, based on the applicant's response, RAI 4.4-27 was resolved.

The staff issued RAI 4.4-28, to request a discussion relating the range of the ATLAS test conditions to expected ESBWR operating conditions and an explanation of the treatment of the electrically heated rods used in the GE14 bundle ATLAS tests that were previously found to influence the result because of magnetic biasing. This RAI requested a discussion of any correction made for this effect. The applicant responded that no adjustment had been made to account for the magnetic biasing attributed to the electrically heated rods of the ATLAS facility and that it would revise Table 4-2 of NEDC-33237P to include the studies that will account for the potential magnetic bias in the ATLAS GE14 critical power data. The applicant submitted the revision of the topical report with no change. In response to RAI 4.4-28 S01, the applicant explained that a conservative adjustment to the GEXL correlation additive constants has been applied to account for the bias in a manner consistent with that applied to operating BWRs and therefore no revision of table 4-2 is required. The staff accepted this explanation, therefore; based on the applicant's response, RAI 4.4-28 was resolved.

The staff issued RAI 4.4-29, to request a discussion of the differences in the tested ranges of: pressure, mass flux, inlet subcooling, and R-factor for a GE14 fuel bundle with regard to the ESBWR operating range. In their response, the applicant proposed a revision to topical report NEDC-33237P, which includes a table indicating that the range of GEXL14 applicability bounds and the corresponding (expected) GE14E conditions for the ESBWR. The staff finds the response to be acceptable, and RAI 4.4-29 was resolved.

The staff issued RAI 4.4-30, to request an explanation of the conservatism of the average experimental CPR, using the adjusted correlation, to the measured critical power from the ATLAS tests. The applicant responded that the uncertainty in the correlation is accounted for by application of an overall correlation uncertainty to ensure conservatism. The applicant added a clarification for Table 4-2 regarding this conservatism. The staff finds the response acceptable. Therefore, based on the applicant's response, RAI 4.4-30 was resolved.

Section 5.14 of NEDC-33237P provides the methodology for determination of the SLMCPR value. The value specified in the TS includes a conservative multiplier to account for the overall GEXL correlation uncertainty. The staff compared the methodology being applied to the ESBWR for consistency with the current operating BWR methodology. The staff finds that the use of the GEXL14 correlation for the determination of the OLMCPR is acceptable.

In RAI 4.4-61, the staff requested the applicant if the GEXL correlation was used to calculate the MCPR in DCD Tier 2, Revision 3, Figures 6.3-7, 6.3-15, 6.3-23, 6.3-31, 4D-22, and 4D-23, which are related to LOCA analyses and startup displayed for these events. If so, the applicant should justify the use of this correlation, since it is being used outside its range of applicability. If not, the applicant should describe the correlation being used and its applicability range.

The applicant responded that the TRACG channel component uses a combination of the GEXL correlation for dryout in annular flow, the Biasi correlation for departure from nucleate boiling, and the Modified Zuber pool boiling CHF correlation for low-flow conditions to determine the transition between nucleate boiling and film boiling. The range of applicability of these correlations is discussed in Section 6.6.6.3 of NEDE-32176. The Modified Zuber correlation is applied below the lower mass-flux limit (100 kilograms per square meter second (kg/m^2-s)) of the Biasi correlation for bubbly or churn flow, which occurs before the transition to annular flow. The TRACG algorithm for determining the “critical power” or “thermal margin” is such that the GEXL correlation is picked for annular flow at higher mass-fluxes (within its range of applicability) and the Modified Zuber or Biasi correlation or their interpolation is picked at lower mass-fluxes (i.e., those outside the range of the GEXL database, but within the range of their applicability). The specific cases of LOCA and startup analyses mentioned in this RAI are discussed below.

LOCA analyses Figures 6.3-7, 6.3-15, 6.3-23, and 6.3-31 of DCD Tier 2, Revision 3, correspond to the feedwater line break, main steamline break, bottom drainline break, and the gravity-driven cooling system (GDCS) injection line break, respectively. Other related figures concerning the static head inside chimney (Figures 6.3-8a, 6.3-16a, 6.3-24a, and 6.3-32a) and peak cladding temperature (Figures 6.3-14a, 6.3-22a, 6.3-30a, and 6.3-38a) show that for all LOCA cases, the ESBWR core is always covered with water and the cladding never heats up. This is consistent with the MCPR values being significantly greater than unity for all LOCA cases, as shown in Figures 6.3-7, 6.3-15, 6.3-23, and 6.3-3 1.

For startup stability analyses, three different heatup rates (50 megawatts (MW), 85 MW, and 125 MW) were used, corresponding to Figures 4D-23 and 4D-24. Initially, there were no voids in the channels, even in the hot bundle. As the heatup proceeds, the RPV pressure (Figure 4D-12) increases; voids start to appear at the exit of the hot bundle (Figures 4D-18, 4D-19, and 4D-20); and the hot bundle exit flow rate (Figure 4D-21) starts to increase. The MCPR is not evaluated until voids appear in the channel or fuel bundle. When the MCPR evaluation starts, TRACG effectively uses the Modified Zuber or Biasi correlation or their interpolation for low mass-fluxes. As the flow rate and pressure increase, TRACG uses the GEXL correlation, as intended, within its range of applicability.

The staff reviewed the submitted information and finds it responsive to RAI 4.4-61, therefore based on the applicant’s response; RAI 4.4-61 was resolved.

The staff presents its detailed evaluation of ESBWR stability in Appendix 4A.
DCD Tier 2, Section 15.2.6, lists the potentially limiting events that establish the OLMCPR as follows:

- loss of feedwater heating (LOFWH) with SCRRI actuation failure;
- slow closure of one temperature control valve;

- generator trip from the 100-percent rated power conditions assuming only 50 percent of the total turbine bypass system capacity; and,

- inadvertent startup of all loops of the isolation condenser system

The staff issued RAI 4.4-62, to request that the applicant should revise topical report NEDC-33237P, Revision 2, Sections 5.12 and 6.0, to reflect the text in DCD Revision 3 regarding the limiting event. In subsequent revisions of DCD Tier 2, Chapter 15, and NEDC-33237P, the limiting events are consistent. Then staff also issued RAI 4.4-62 S01 to suggest that the same inconsistency for the Loss of Feedwater heating with SCRRI failure and the Inadvertent Isolation Condenser Initiation (IICI). The limiting events were altered as a result of design changes, which include addition of the feedwater temperature operating domain, and changes in the component flow loss coefficient resulting from core support plate and other dimensional changes. The OLMCPR for each fuel cycle will be established for the limiting event and documented in the core operating limits report (COLR) in accordance with the TSs. The staff finds the applicant's response acceptable, therefore, considers RAI 4.4-62 and RAI 4.4-62 S01 as resolved.

NEDO-33337, "ESBWR Initial Core Transient Analysis," issued October 2007, provides the analysis of these events for the initial core. The staff discusses its evaluation in Section 15 of the FSER. Reanalysis of these events will be performed for reload core designs. The results shall be reported in the COLR, as specified in Section 5.6.3(a) (2) of the ESBWR TS (DCD Tier 2, Chapter 16).

4.4.3.4 Void Fraction

The ESBWR is expected to operate at up to approximately 90 percent void for normal steady state operation and AOOs. This is significantly higher than for conventional BWRs. To assess the adequacy of the void fraction correlation used by the applicant the staff issued RAI 4.4-2, and requested the database used to develop the correlation. In its response, the applicant referred to the staff approved licensing topical report NEDE-21565. The response also provided validation data for the expected range of ESBWR operating conditions for simple test geometries and various fuel geometries, including 4x4, 6x6, 7x7, and 8x8. The staff reviewed the relationship between the nodal void and nodal quality and made the following observations regarding the void-quality correlation topical report:

- The report, prepared in 1977, includes data for 4x4, 6x6, 7x7, and 8x8 fuel bundle designs. It does not include test data for newer fuel designs with greater than 8x8 bundles (such as the 10x10 arrangement of the ESBWR fuel). The uncertainty in the correlation related to geometry effects should be addressed for the newer fuel designs.

- Most of the test data were concentrated at approximately normal BWR operating pressure (1,000 psia). A few measurements were taken at lower or higher pressures (400 to 1,400 psia) for the various bundle geometries. The void fraction correlation is based on an extensive database for the expected normal operating pressures and flow rates. Outside the normal range, there is a significant uncertainty associated with extrapolation of the correlation to high or low void fractions.

- The correlation is biased downward by a factor of two weighting with the CISE (Ciencias de la Seguridas test facility) 4x4 fuel bundle data, since the quick-closing valve arrangement of the CISE tests was considered most reliable. Expected differences in results for 10x10 bundles should be addressed.

- No data are available for the counter-current flow regime. Test data should be acquired for this regime, or justification should be provided for not considering this flow regime.

- No transient testing was performed. Transient data should be acquired to confirm the void fraction correlation accuracy in transient conditions, or justification should be provided for not considering transient conditions.

- Some of the test data were skewed by radial peaking of the power distribution. Additional full-scale void fraction data with skewed radial peaking should be acquired, or justification should be provided for not considering this effect.

The staff found several areas of uncertainty in applying the Findlay-Dix void quality correlation to new GEH fuel designs at high void fraction. The staff also evaluated the formulation of the correlation. The correlation is based on a two-fluid semi-empirical model. The staff is not certain of the appropriateness of the correlation for predicting void fraction above its originally qualified range for new designs. Specifically, the staff is uncertain about the ability of the model to adequately account for the effects of entrained liquid droplets in the vapor core for high void fractions, where the liquid droplets represent an increasingly large fraction of the liquid flow. In its response to RAI 4.4-2, the applicant referenced the approved TASC-03A code in topical report NEDC-32084P--A, Revision 2, and the approved GESTAR topical report described in MFN 003-086, "Revised Supplementary Information Regarding Amendment 11 to GE Licensing Topical Report NEDE-24011-P-A," dated January 16, 1986. Section 3.1 of NEDE-32177P, Revision 2, discusses the TRACG code qualification of the void fraction determination approach.

In a follow up the staff issued RAI 4.4-2 S02 regarding the applicant's response to RAI 4.4-2, the staff indicated that the indirect justification provided in the supplemental response for the void correlation at high void fractions using operating fleet GE14 pressure drop data is not a substitute for actual void fraction measurements. Additionally, the staff proposes to apply the same adder to the OLMCPR imposed on the GE14 fuel (MFN 06-435, dated November 3, 2006, and Supplement 1 to NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains") as a penalty to account for the uncertainty in the void fraction correlation resulting from the lack of high void fraction data until the data are acquired and assessed. The staff requested the applicant to revise NEDC-33237P to document the proposed OLMCPR penalty. The approved version of NEDC-33237P will include a discussion of the imposed thermal margin adder. The SER for NEDC-33237P includes the OLMCPR adder as a condition for approval of the calculation methodology, which must be satisfied for licensing applications. The staff considered the OLMCPR adder a satisfactory response and therefore, considers RAI 4.4-2 and RAI 4.4-2 S01 as resolved.

The applicant provided additional qualification data to demonstrate that the range of expected operating void fractions (from 0 percent to 92 percent) for the ESBWR is within the qualification basis of the void fraction methods. To account for the uncertainties in void fraction prediction for 10x10 fuel bundles, the staff considers it necessary to apply an OLMCPR adder as a penalty.

Because the void reactivity coefficient is a strong function of the void fraction (increasing in magnitude with increasing void fraction), and given the specific concerns regarding the void quality correlation listed above and concerns about the efficacy of the core simulator code (PANACEA) in producing reliable nuclear data for use in downstream transient analysis codes where void fractions may exceed 90 percent locally, the staff approval of the PANAC11 methodology for the ESBWR will be contingent on an additional margin to the ΔCPR in the OLMCPR determination. An adder of 0.01 to the OLMCPR is consistent with an approximately 0.5-percent additional uncertainty in nodal transient power. The staff will require that an adder of 0.01 be in place for the ESBWR OLMCPR methodology until the capabilities of the Findlay-Dix correlation are demonstrated for modern fuel designs over the range of void expected for steady-state operation and AOOs characteristic of the ESBWR. Additional detailed discussion of the safety evaluation is presented in NEDC-33239P Revision 4, and NEDE-33197P Revision 4.

4.4.3.5 Core Pressure Drop and Hydraulic Loads

To evaluate the method, assumptions, and results used by the applicant to calculate core pressure drop and component hydraulic loads, the staff requested, in RAI 4.4-20, a discussion of the calculation of the reactor internal pressure drop and associated loads for normal and transient operation. The applicant responded that the TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs, infrequent events, and accidents (e.g., LOCAs).

The discussion in DCD Tier 2, Section 3.9.5, pertains to reactor internal components. The fuel assembly, including the fuel rods, is not considered a reactor internal; however, the pressure differences determined in the section are also used to evaluate the hydraulic loads on the fuel assembly. DCD Tier 2, Section 4.2.3, discusses the hydraulic loads and the resulting stresses for the fuel channel. Details of this analysis appear in Section 3.4.1.8 of LTR NEDC-33240P.

The reactor internal pressure differences are calculated by appropriate selection in the TRACG model of axial and radial nodes, connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. Component hydraulic loads are calculated using the transient reactor internal pressure difference and the projected area of the component.

Section 21.6 of this report presents the staff's assessment of the TRACG program for the ESBWR design. The staff has previously approved the LTR on TRACG application for the ESBWR, NEDE-33083P-A, Revision 0, for ESBWR LOCA application. The staff has also approved TRACG for the ESBWR stability analysis, and approval of TRACG for the ESBWR AOO application is part of the ESBWR design certification application. The staff review of NEDE-33083P-A, Revision 0, also includes the TRACG model description topical report, NEDE-32176, by reference. The staff finds the above discussion acceptable and responsive to the request, therefore, RAI 4.4-20 was resolved.

In RAI 4.4-24, the staff requested further discussion of the pressure drop qualification test data used to develop pressure loss coefficients. In its response, the applicant stated that the GE14E fuel design uses hardware identical to that currently used in GE14 fuel assemblies. Therefore, the component local pressure drop characteristics will be the same. Topical report NEDC-33238P provides test results for the GE14 components at various flow rates and power levels. The range of test conditions includes the expected ESBWR operating range. The staff finds this acceptable.

As discussed in Section 21.6 of this report, core pressure drop testing was ranked high on the phenomena identification and ranking table (PIRT) for ESBWR AOOs. The staff finds the application of TRACG for the determination of core pressure drop and hydraulic loads acceptable, provided that the confirmatory items identified in Section 21.6 are satisfied, therefore, based on the applicant's response, RAI 4.4-24 was resolved.

4.4.3.6 Core Coolant Flow Distribution

The staff issued RAI 4.4-23, to request a quantitative comparison of pressure drops and flow distributions in the fuel channels and core bypass regions of the ESBWR to those of conventional BWRs, as well as a discussion of the impact of a flow reduction on the MCPR limit. In its response the applicant provided a table comparing core pressure drop, bypass flow, and fuel channel flow characteristics of the ESBWR to a BWR/6 and a BWR/4 plant of conventional design, with recirculation pumps. The ESBWR core diameter is similar in size to that of a BWR/6.

In DCD Tier 2, Tables 4.4-1a and 4.4-1b provide typical thermal-hydraulic design characteristics of the reactor core. The ESBWR design parameters are compared to those of the conventional BWR/6 and the ABWR. The data reflect the differences expected because of the natural circulation design of the ESBWR. The staff finds the comparison responsive to the request and acceptable; therefore, RAI 4.4-23 is considered resolved.

In addition, the staff issued RAI 4.4-23 S01, to request a calculation of MCPR as a function of percent of flow blockage. The applicant provided the results of the flow blockage calculation in letter MFN 06-399. The calculation shows that a significant portion of the initial flow area for the inlet orifice or for the initial flow area of the lower tie plate must be blocked before boiling transition (CPR = 1.0), is reached. The ESBWR lower plenum velocities are lower than those in forced circulation BWRs, which should reduce the chance that foreign material is swept up to the inlet orifice or lower tie plate. The lower velocity also minimizes impingement of debris on the bundle fine screen filters. The staff finds the response acceptable in view of the significant blockage required to cause the core to reach boiling transition, therefore, RAI 4.4-23 S01 was resolved.

The staff issued, to request the list of assumptions made in the calculations presented in the response to RAI 4.4-23 S01. The applicant provided the list of assumptions made in the calculations. The staff reviewed the list and finds the assumptions to be reasonable and the calculation approach conservative. The response provided justification for the limited debris quantity, types, and sizes that can be expected during an ESBWR LOCA.

The primary debris pathway from the containment to the reactor internals, the opening above the GDCS pool, is protected by a perforated steel plate. Also, the natural circulation coolant flow velocity will be much lower than that of conventional forced circulation BWR, so the amount of debris entrainment is significantly less than expected for conventional BWRs. The staff finds the response acceptable and RAI 4.4-23 S02 to be resolved.

Regarding the debris entrainment issue the applicant committed to addressing compliance with RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident.” In addition, the response to RAI 6.2-173 S01, discusses this topic. The safety evaluation of DCD Tier 2, Section 6.2, presents the staff’s evaluation of the debris transport.

In RAI 4.4-23 S03, the staff requested the applicant to address similarities and differences between the ESBWR fuel and cooling systems and to address the applicability of the BWR Owners Group (BWROG) calculations for downstream effects of LOCA-generated debris. The Owners Group calculations demonstrate that acceptable fuel centerline temperatures can be maintained. The ESBWR thermal insulation is limited to the stainless steel reflective metallic type, which significantly minimizes the quantity of debris that can be transported to the reactor vessel. Operating BWR containments include significant quantities of fibrous insulation material, which can contribute to debris blockage. The ESBWR design ensures that the reactor vessel water level can be maintained above the top of active fuel for any postulated LOCA. Even if one or more fuel bundle inlet orifices are blocked, the channels remain water filled, with flow coming from the top.

In RAI 4.4-23S04, the staff requested a calculation similar to that performed for the BWROG. The staff’s concern was that, during the early portion of the transient when the decay heat remains high and rapid outflow of liquid inventory occurs as a result of depressurization of the reactor from the pipe rupture, some fuel rods may lose cooling and fuel damage may occur. The applicant performed TRACG calculations that demonstrate that saturated liquid conditions can be maintained at all times during a LOCA. Substantial thermal margin is calculated. Section 6.3 of this SER summarizes the staff evaluation of ECCS performance for a spectrum of postulated line breaks, including the effects of debris blockage. Based on the applicant’s responses above RAI 4.4-23 S03 and RAI 4.4-23 S04 are resolved.

4.4.3.7 Fuel Heat Transfer

The staff issued RAI 4.4-3, to request a discussion of the heat transfer bases. The applicant responded that standard and well-accepted heat transfer correlations between the coolant and the rod surfaces are used. Topical reports NEDE-32176P and NEDE-33083P-A describe these correlations in detail. Section 21.6 of this report presents the staff’s assessment of the TRACG program heat transfer model for the ESBWR design that has been reviewed and approved by the staff, therefore, based on the applicant’s response, RAI 4.4-3 was resolved subject to the conditions and limitations listed in the LTR.

4.4.3.8 Maximum Linear Heat Generation Rate (MLHGR)

The MLHGR is the maximum local heat generation rate (more specifically, the fuel rod with the highest surface heat flux at any nodal plane in a fuel bundle in the core). The MLHGR operating limit depends on the bundle type, and Section 4.2 of this report evaluates the determination of this limit. ESBWR TS 3.2.1 specifies the LHGR. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel will not occur during any AOOs. The SER for NEDC-33239P provides a detailed evaluation of the linear heat generation rate (LHGR) determination as a function of core power distribution, which is dependent on the nuclear flux shape. The staff finds the applicant's method for determination of LHGR limits acceptable.

4.4.3.9 Core Power/Flow Operating Map

As stated in RAI 4.4-63, the staff noted that in DCD Tier 2, Revision 3, Section 4.4.4.3, the applicant added a statement that a core power-feedwater temperature operating map is envisioned. Previous revisions of the DCD had stated that the core power-flow map is only a single line, and there is no active control of the core flow at a given power level. In addition, DCD Tier 2, Revision 3, Section 4.4.4.4, "Temperature-Power Operating Map," currently states, "Not Applicable to the ESBWR." The staff expects that the applicant will revise this section to reflect the change to a temperature-power operating map. In NEDO-33338, the applicant provided additional information on the proposed use of feedwater temperature variations to maneuver reactor power. The applicant also added Figure 4.4-1, "Typical ESBWR Core Power—Feedwater Temperature Operating Domain/Map." Chapter 15 of the FSER discusses the staff evaluation of NEDO-33338 that has been reviewed and approved by the staff. Subsequent DCD revisions incorporate changes to the text in Section 4.4.4.3, which satisfactorily address the staff's concerns stated in RAI 4.4-63, therefore, based on the applicant's response and the staff approval of NEDO-33338, RAI 4.4-63 was resolved.

4.4.3.10 Inadequate Core Cooling Monitoring System

The staff issued RAI 4.4-21, to request a description of the ESBWR "Inadequate Core Cooling" (ICC) monitoring system to satisfy the requirements of SRP Section 4.4. The applicant responded by providing an additional section to the DCD that refers to DCD Tier 2, Table 1A-1 (TMI Action Plan Item II.F.2). The staff reviewed the revised table and finds it acceptable with respect to the thermal-hydraulics detection capability of the system. Chapter 7 discusses the instrumentation and control room display aspects of the system. Additional staff evaluation of the ICC system appears in Section 20.3 of this report. Based on the applicant's response, the added section in the DCD and Table 1A-1, RAI 4.4-21 was resolved.

4.4.3.11 Loose Parts Monitoring System (LPMS)

The staff issued RAI 4.4-7 to RAI 4.4-9 (and corresponding S01s) requesting GEH provide information regarding the implementation and operation of the loose parts monitoring system. In response to RAIs 4.4-7 through RAI 4.4-9, (and corresponding S01s) the applicant informed the staff that intends to delete the LPMS from the ESBWR design and provided a basis for doing so. In addition, it stated that "small metallic filings and other similar debris could contribute to fuel cladding damage, but the LPMS would not detect this class of debris, and the industry has installed debris filters into the fuel support pieces which may reduce fuel cladding damage due to fretting." The applicant further noted that the ESBWR design incorporates debris filters and that all fuel supplied by the applicant has a filter (at the bottom) to prevent debris from entering the bundle.

The staff also issued RAIs 4.4-7 S02 through RAI 4.4-9 S02 to request supplemental information to assist the staff in determining whether deletion of the LPMS from the ESBWR design is acceptable. The request included a detailed discussion of: (1) the design of ESBWR debris filter, (2) the maximum size of debris that can pass through the filter, and (3) adverse impacts on cladding and other components in the core caused by the debris that passes through the filter. Additionally, the staff requested an assessment of the adverse impact on ESBWR safety-related systems and components caused by the debris that originates downstream of the filter. The safety assessment should address the potential for physical damage and flow blockage, particularly focusing on the ESBWR's unique features, including the potential for flow blockage of natural circulation and gravity-driven flow lines.

The staff specified that the response was to include, but not be limited to, the following ESBWR components:

- depressurization valves (DPVs);
- main steam isolation valves (MSIVs);
- isolation condenser system—tubes and valves;
- SLCS—injection lines;
- GDCS—injection lines and valves; and,
- CRD system.

The staff also requested a detailed explanation and demonstration that the ESBWR can be safely operated without an LPMS. A systematic analysis of all systems and components in the RPV and the connected systems is required to justify the deletion of the LPMS.

In response to the supplemental RAIs, the applicant stated that the same debris filters used in GE12 fuel are integrated into the lower tie plates of each fuel bundle in the ESBWR. Water must pass through the flow holes before entering the fuel bundles. The LPMS would not detect objects small enough to pass through the filters.

Additionally, the applicant indicated that it is expected that licensees will employ a rigorous foreign materials exclusion program to prevent external sources of loose parts. They will also conduct underwater visual vessel internals inspections during outages to check the structural integrity of reactor components. This will also provide opportunities to find loose parts in the area where inspections are performed. In RAI 4.4-9 S02, the staff requested that the applicant incorporate into the DCD the justification for not providing an LMPS for the ESBWR. In addition, the staff requested an update to DCD Table 1.9-21 for RG 1.133, indicating that the LPMS will be deleted from ESBWR design. The applicant responded with proposed revisions to the DCD as requested in the RAIs, therefore, based on the applicant's responses, RAIs 4.4-7, 4.4-8, and 4.4-9 (and corresponding S01 and S02) are resolved.

The ESBWR design considers important aspects such as material selection and analysis for internal components to prevent failures, and it uses proven design methods to fasten components. In addition, in accordance with RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals during Pre-Operational and Startup Testing,” the applicant plans to install instruments on components during initial startup of the lead ESBWR plant as part of a program to measure the flow-induced vibration of critical components. The type and location of instrumentation is established by detailed evaluations of the RPV components, using prior test data and analyses to determine susceptibility to flow-induced vibration. This minimizes the possibility for internal sources of loose parts in the RPV because of vibration.

The only two systems that provide fluid flow directly into the RPV during normal operation are the feedwater and CRD system. The feedwater system uses temporary strainers as a precautionary measure to ensure that loose parts that may have been left during the construction phase do not enter the RPV. These temporary strainers and any debris collected are removed after the first cycle. Also, the feedwater sparger inside the RPV provides a difficult path for large objects to pass through and enter into the RPV. Objects entering the feedwater spargers must pass through a 2-inch short-radius elbow and then pass through a nozzle with a maximum diameter of 1.875 inches to enter the RPV. Objects that are restricted within the feedwater spargers do not adversely affect the operation of the plant or the feedwater spargers. This minimizes the opportunity for loose parts to enter the RPV through the feedwater system.

For the CRD system, purge water flow enters from the bottom of the FMCRD through a 1.25-inch line. Because of the restricted flow paths within the drive, only small objects that an LPMS would not detect could possibly enter the drive. GEH concurs with the Electric Power Research Institute (EPRI) assessment that a loose part is not likely to enter into the CRD and restrict its operation. NEDC-32975P-A, Revision 0, “Regulatory Relaxation for BWR Loose Parts Monitoring Systems,” issued February 2001, states the following:

> The EPRI report also stated that loose parts do not, in general, affect CRD operation, because of the torturous path required for loose parts to enter the CRD guide tube. From the upper plenum, the clearance between the fuel channel and the top of the guide tube is small and movement of any loose parts would be counter to core flow. From the lower plenum, access to the CRD guide tube by metallic parts is effectively prevented by the integrity of the guide tube and the core flow patterns that exist in the fuel bundle and bypass regions. Any debris which enters a CRD guide tube is unlikely to have sufficient mechanical strength to interfere with the operation of the CRD.

In the safety evaluation of the topical report NEDC-32975P-A, the staff also agreed with the EPRI report’s evaluation that small loose parts or debris from the lower plenum will probably not impede CRD operation because of the difficult flow path. The staff further stated that small loose parts and debris could enter the CRD during refueling, but the LPMS will not likely detect this class of debris.

The applicant stated that in the event of a loose part entering the vessel, the ESBWR design is capable of performing its safety-related functions. The plant has been designed with multiple DPVs and safety/relief valves (SRVs). In the event a DPV or SRV is restricted, the remaining DPVs and SRVs can accomplish the task of blowdown. The plant has been designed with redundant MSIVs. If one of the series MSIVs becomes restricted, the remaining MSIV can accomplish the task of isolation. The isolation condenser system has four independent trains. If one of the trains is restricted, the remaining three trains can accomplish the task of heat removal. The SLCS has two independent trains. Each train has an injection line that branches into two sets of three injection nozzles within the core shroud. If one of the injection nozzles becomes restricted, the remaining 11 nozzles can accomplish boron injection. The GDCS has four independent trains. If one of the trains is restricted, the remaining three trains can accomplish the task of supplying inventory for a LOCA. Design and testing are performed appropriately to ensure that loose parts are not generated internally. Foreign materials exclusion programs are performed to limit externally generated loose parts from entering the reactor coolant pressure boundary (RCPB). Underwater in-vessel visual inspections are performed to detect cracking of components that can become potential loose parts. In addition, with the redundancy in the design of the safety systems, GEH concluded that the ESBWR is capable of performing its safety-related functions without an LPMS. The staff concurs with the GEH justification described above for not including the LPMS in the ESBWR design.

It may also be noted that the staff agreed with the BWROG regarding the deletion of the LPMS from the currently operating plants. The staff stated that the safety benefits of the LPMS do not appear to be commensurate with the safety benefit and the associated radiation exposure of plant personnel.

4.4.3.12 Testing and Verification

The staff reviewed the ITAAC listed in DCD Tier 1, Table 2.1.1-3, pertaining to the RPV and internals. These ITAAC are intended to ensure that the as-built component dimensions and arrangement are consistent with the design analyses. The staff also reviewed DCD Tier 1, Table 2.1.2-3, "ITAAC for the Nuclear Boiler System," for impact on core thermal-hydraulic design. Parameters that have been used in the design analyses for natural circulation flow, such as pressure loss coefficients, component free volumes, geometry, hydraulic diameters, and flow areas, will be confirmed in the as-built reactor vessel before fuel loading. The staff finds the proposed ITAAC appropriate and complete.

4.4.4 Conclusions

The applicant has met the requirements of GDC 10 and 12 with respect to SAFDLs by providing analyses and test results demonstrating that normal operation, including the effects of AOOs, will satisfy the fuel design criteria, provided that the conditions and limitations applicable to approved topical reports are satisfied. These topical reports describe the methods and assumptions used for the evaluation of the reactor thermal and hydraulic design. DCD Tier 2, Section 6.3, presents analyses related to core thermal and hydraulic design for emergency core cooling, and DCD Tier 2, Chapter 15, presents the transient and accident analyses. The corresponding FSER sections present the staff evaluation. DCD Tier 2, Appendix 4D, and FSER Appendix 4A specifically addresses GDC 12.

4.5 <u>Reactor Materials</u>

4.5.1 Control Rod Drive System Structural Materials

The staff reviewed DCD Tier 2, Section 4.5.1, “Control Rod Drive System Structural Materials,” in accordance with SRP Section 4.5.1, “Control Rod Drive System Structural Materials.” The CRD structural materials are acceptable if the relevant requirements of the following regulations are met:

a. 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.

b. 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.

c. GDC 26, "Reactivity Control System Redundancy and Capability" 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.

Descriptive information on the FMCRD, as well as the entire CRD system, appears in DCD Tier 2, Section 4.6.1. As described below, the staff reviewed the structural materials aspects of the CRD, as presented in the DCD, in accordance with the guidelines in SRP Section 4.5.1.

4.5.1.1 Summary of Technical Information

DCD Tier 2, Section 4.5.1, describes the materials used to fabricate structural components of the CRD system. The DCD also provides information about the materials specifications, the fabrication and processing of austenitic stainless steel components, the contamination protection and cleaning of austenitic stainless steel, and items concerned with materials other than austenitic stainless steel.

The metallic structural components of the CRD mechanism are fabricated from four types of materials, which include 300 series stainless steel, nickel-chromium-iron (Ni-Cr-Fe) Alloy X-750, XM-19, and 17-4 PH materials. The primary pressure boundary components of the CRDs are the lower housing of the spool piece assembly, the flange of the outer tube assembly, and the mounting bolts. The applicant stated that all materials used in the CRD system are selected for their compatibility with the reactor coolant as described in Sub-articles NB-2160 and NB-3120 of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter referred to as the ASME Code).

Pressure-retaining materials comply with the ASME Code, Section III, which 10 CFR 50.55a, “Codes and Standards,” incorporates by reference. DCD Tier 2,

Table 5.2-4, includes the materials specifications for portions of the CRDs that are part of the RCPB and are fabricated from forged austenitic stainless steel (Type F316/F316L and Type F304/F304L).

The CRD system does not employ austenitic stainless steels strengthened by cold work. For incidental cold work introduced during fabrication and installation, special controls are used to limit the induced strain and hardness, and the bend radii are kept at a minimum value.

Stellite 3/Haynes 25 is used for rollers/pins at latch (outside), and Haynes 25 is used for the latch joint pin. A material equivalent to Stellite 6 is used in the guide shaft at the top of the ball spindle. Stellite 12 is used for the bushing at the top of the ball spindle and the bushing on the drive shaft. Stellite Star J-metal is used for the ball check valve. Noncobalt hard surfacing alloys are used in guide rollers and guide pins. These components are located above and below the labyrinth seal and on the stop piston, ball screw stationary guide, piston head, and ball nut.

4.5.1.2 Evaluation

The staff reviewed and evaluated the information in DCD Tier 2, Revision 1, and Section 4.5.1, to ensure that the materials specifications, fabrication, and process controls are in accordance with the criteria of SRP Section 4.5.1.

4.5.1.2.1 Materials Specifications

The staff reviewed DCD Tier 2, Section 4.5.1.1, “Materials Specifications,” to determine the suitability of the materials for this application. The DCD 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 middle flange, the spool piece, and the mounting bolts. The middle flange and spool piece components are fabricated from austenitic stainless steel forgings (SA-336 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 ESBWR design.

The remaining components identified in DCD Tier 2, Section 4.5.1, are not RCPB materials. The DCD 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, “Materials Code Case Acceptability—ASME Section III, Division I,” and are therefore acceptable for use in CRD components.

4.5.1.2.2 Austenitic Stainless Steel Components

The applicant indicated that all stainless steel materials are used in the solution heat-treated condition. For all welded components exposed to service temperatures exceeding 93 degrees C (200 degrees F), 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 degrees C (200 degrees F) 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," issued January 1988, which is consistent with SRP Section 4.5.1. The applicant indicated that significantly cold-worked 300 series austenitic stainless steels are not used. 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. In RAI 4.5-31, the staff requested the applicant to provide the values of the ESBWR design special controls limits on hardness, 0.2-percent offset yield strength, and induced strain. The staff also requested the applicant to discuss the abrasive work controls for limiting cold working and the introduction of contaminants during abrasive processes. Finally, the staff requested the applicant to provide its response in a generic sense as it applies to the entire ESBWR design.

In response, the applicant stated the following:

> GEH applies special cold work controls to all stainless steel in the reactor system, defined as components inside containment continuously exposed to reactor water greater than 93 °C (200 °F). Bulk hardness of all stainless steels in the final fabricated condition (with the one exception noted in the response to RAI 4.5-12) is controlled to Rockwell B-90 for Types 304/304L and Rockwell B-92 for Types 316/316L. Cold forming and straightening strains are limited to 2.5 percent, or alternately, in the case of bars, plate, or pipe, a bend radius greater than 20 d or t (diameter or thickness). Additionally, for the major structural welds of core support structures and large internal components, polishing of the weld heat affected zones is required to remove surface cold work introduced by forming, machining, or grinding. Maximum yield strength is not controlled specifically, but the combination of solution heat treatment controls, hardness controls, and cold forming controls assure that, in all cases, the yield strength of stainless steels is far below 90,000 psi. Grinding is controlled by requiring ground areas to be polished to remove surface cold work introduced by grinding. Grinding media are controlled by requirements that processing materials shall be low in halogens, sulfur, and low melting point metals as well as thorough final cleaning of all ground surfaces. Additionally, it is required that grinding media be new, or previously used only on stainless steel or nickel alloys.

The maximum hardness limit as specified by the applicant is consistent with the acceptance criteria specified in SRP Section 4.5.1. The staff considers the applicant's special 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's response is acceptable. Based on the applicant's response, RAI 4.5-31 was resolved. DCD Tier 2, Section 4.5.1.2.1, states that Section 4.5.2.2 discusses the degree of conformance to RG 1.44, "Control of the Use of Sensitized Stainless Steel." In Section 4.5.2.2, the applicant indicated that the ESBWR design complies with the intent of 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. In RAI 4.5-29, the staff requested the applicant to clarify its compliance with the guidelines of RG 1.44 because its use of the word "intent" does not make it clear whether the ESBWR design is consistent with all of the guidelines in RG 1.44. The staff also requested the applicant to specify the type of test it will use to detect susceptibility to intergranular attack in austenitic stainless steels in the ESBWR design in order to conform to the guidance in RG 1.44.

In response, the applicant clarified that it used the word “intent” in a general sense and indicated that sensitized stainless steel will not be used. The test used to detect susceptibility to intergranular attack is a modified version of American Society for Testing and Materials (ASTM) A 262, “Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels,” Practice A, that more strictly defines rejectable ditching and that does not allow retest and acceptance under Practice E. The applicant also indicated that the ESBWR design will comply with RG 1.44. 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. Based on the applicant’s response, RAI 4.5-29 was resolved.

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. DCD Tier 2, Section 4.5.2.2, indicates that the staff guidance in RG 1.31 will be followed, which is acceptable.

4.5.1.2.3 Other Materials

The DCD identifies that the bayonet coupling, latch and latch spring, and separation spring (non-RCPB components) are fabricated from Alloy X-750 in the annealed condition and aged 20 hours at 1,300 degrees F. In RAI 4.5-28, the staff requested that the applicant discuss the relationship between the thermal and mechanical processing of Alloy X-750 components and their susceptibility to SCC.

In response, the applicant referred to its response to RAI 4.5-13 regarding Alloy X-750. In that response, the applicant indicated that the heat treatment for Alloy X-750 components, other than a spring on a shroud head bolt or a latch component in the steam dryer, is consistent with the Type 3 heat treatment of ASTM/ASME B/SB-637, “Specification for Precipitation-Hardening Nickel Alloy Bars, Forgings, and Forging Stock for High-Temperature Service” and the EPRI guidelines on X-750. The high-temperature anneal treatment in conjunction with a single step aging treatment is considered to provide optimum stress-corrosion resistance in X-750 in BWR applications. The applicant also indicated that, although it is believed that hardness in excess of Rc40 can indicate elevated susceptibility to SCC, B/SB-637 Type 3 heat treatment specifies an Rc40 maximum hardness. Based on industry experience using X-750 in CRD components, the use of EPRI heat treatment guidelines, and the accessibility of these components for inspection and replacement if necessary, the staff finds the applicant’s use of Alloy X-750 for CRD components acceptable. Based on the industry’s experience reported in the applicant’s response, RAI 4.5-28 was resolved.

The CRD ball spindle and ball nut are fabricated from 17-4 PH stainless steel in condition H-1075 (aged 4 hours at 1,075 degrees F). SRP Section 4.5.1 identifies 1,075 degrees F as an appropriate aging temperature for CRD components fabricated from 17-4 PH stainless steel. Therefore, the staff finds the applicant’s heat treatment of 17-4 PH acceptable.

In the CRDs, cobalt-bearing and non-cobalt-bearing alloys are specified for wear and hard surfacing applications. Radiation buildup during plant operation can occur because of cobalt-60, which forms by neutron activation of cobalt-59. In RAI 4.5-27, the staff requested the applicant to discuss the basis for selection, operating experience with the materials selected, and use of cobalt-bearing and non-cobalt-bearing wear-resistant alloys in the ESBWR design.

In response, the applicant stated the following:

> Other than the cobalt bearing materials in the FMCRD noted in DCD Section 4.5.1, no cobalt bearing alloys are used in the ESBWR internals design. The components in the FMCRD are small bearings and other parts where maximum wear resistance is required. Because these materials are contained within the CRD, they are not directly activated because of being located far below the bottom of active fuel where neutron fluence is minimal. Release of cobalt to reactor water by general corrosion is very limited because the operating temperature inside the drive is substantially lower than reactor temperature, flow rates are low, and these cobalt base alloys have generally high corrosion resistance. The non-cobalt alloys used in wear and hard surfacing applications in the FMCRD components were selected specifically to minimize the use of cobalt base alloys. These alloys were qualified for the FMCRD application by extensive mockup testing for ABWR and have been in service in Kashiwazaki-Kariwa 7 since it started up in 1997. Any of these components are readily replaceable as part of routine CRD maintenance.

The staff finds the applicant's response acceptable, given that direct activation of cobalt-bearing alloys is unlikely and release of cobalt because of general corrosion is limited. Extensive mockup testing and a service history using these alloys for FMCRD components provide assurance of the capability of these materials to perform their intended function. Based on the above discussion in the applicant's response, RAI 4.5-27 was resolved.

4.5.1.2.4 Compatibility of Materials with the Reactor Coolant

The materials selected for use in the CRD system must be compatible with the reactor coolant, as described in Subarticles NB-2160 and NB-3120 of the ASME Code, Section III. The information in the DCD indicates that the RCPB materials used in the CRD system are compatible with the reactor coolant and, thus, comply with the ASME Code, Section III, Subarticles NB-2160 and NB-3120.

Furthermore, the materials selected for the CRD system are currently in use in nuclear power plants and have been proven to perform satisfactorily under the environmental conditions found in these plants. The staff finds the selected materials for this element of design to be acceptable because they perform satisfactorily under the expected environmental conditions.

4.5.1.2.5 Cleaning and Cleanliness Control

The staff's acceptance criteria for cleaning and cleanliness controls conform to RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated

Components of Water-Cooled Nuclear Power Plants." The ESBWR design conforms to RG 1.37, with the exception of quality standard American National Standards Institute (ANSI) N45.2.1, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," 1973 referenced in RG 1.37. DCD Section 4.5.1.4 references Part 2.2 of NQA-1, "Quality Assurance Program Requirements for Nuclear Power Plants," and RG 1.37. In RAI 4.5-30, the staff asked the applicant to specify the edition of NQA-1 that is applicable. The staff notes that the ESBWR DCD references NQA-1-1983, in Chapter 17, but the applicable section related to the requirements for cleaning of fluid systems and associated components is located in NQA-2-1983, "Quality Assurance Requirements for Nuclear Facility Applications." The staff requested the applicant to provide clarification and state whether all positions of RG 1.37 are being met in a global context as it applies to the entire ESBWR design.

In response, the applicant stated the following:

> The ESBWR design commitment in DCD Table 1.9-22 will be changed to NQA-1-1983 and NQA-2-1983 in response to NRC review of DCD Chapter 17. All references in the DCD to NQA-1 and/or NQA-2 will be revised accordingly. The ESBWR design complies with RG 1.37 except as noted in DCD Table 1.9-21b. The NRC has accepted an alternate position as documented in Table 2-1 of DCD Reference 1.9-2 (GEH Nuclear Energy Quality Assurance Program Description, March 31, 1989, NEDO-11209-04a, Class I (non-proprietary) Revision 8). The alternate position is stated as follows: "Comply with the provisions of Regulatory Guide 1.37, March 16, 1973, including the requirements and recommendations in ANSI N45.2.1-1973, except as follows:
>
> "Section 5, sixth paragraph, recommends that local rusting on corrosion resistant alloys be removed by mechanical methods. This recommendation shall be interpreted to mean that local rusting may be removed mechanically, but that it does not preclude the use of other removal means. In addition, the ESBWR design complies with the cleaning requirements of ANSI N45.2.1-1980 and the packaging, shipping, receiving, storage and handling requirements of ANSI N45-2.2-1978 as referenced in DCD Table 1.9-22. Compliance is met by means of their incorporation into NQA-2-1983. DCD Section 4.5 will be revised in the next update to specify NQA-2-1983, Part 2.2 in Subsection 4.5.1.4 instead of NQA-1, Part 2.2."

The staff verified that the applicant made the above-cited changes to the DCD. Given that the NRC staff has previously approved the use of NQA-1-1983, NQA-2-1983, and the applicant's alternative, the staff finds that the applicant meets the guidelines provided in RG 1.37, and its position is, therefore, acceptable. Based on the above discussion and the applicant's response, RAI 4.5-30 was resolved.

4.5.1.3 *Conclusions*

The staff finds 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, 14, and 26, as well as 10 CFR 50.55a.

4.5.2 Reactor Internal Materials

The staff reviewed ESBWR DCD Tier 2, Section 4.5.2, “Reactor Internal Materials,” in accordance with SRP Section 4.5.2, Revision 3, “Reactor Internal and Core Support Materials,” issued March 2007. The design, fabrication, and testing of the materials used in the reactor internals and core support structures are acceptable if they meet codes and standards commensurate with the safety functions to be performed. The acceptability of the materials will ensure that the relevant requirements of 10 CFR 50.55a and GDC 1 are met. The following specific acceptance criteria of SRP Section 4.5.2 are necessary to meet the requirements of 10 CFR 50.55a and GDC 1:

- Materials Specifications, Selection, and Heat Treatment

  For core support structures and reactor internals, ASME Code, Section III, Subarticle NG-2000, identifies and describes the permitted materials specification. ASME Code cases approved for use identify additional permitted materials and their applications, as described in RG 1.84.

  All materials used for reactor internals and core support structures must be compatible with the reactor coolant, as described in ASME Code, Section III, Subarticles NG-2160 and NG-3120. The tempering temperature of martensitic stainless steels should be specified to provide assurance that these materials will not deteriorate in service.

- Controls on Welding

  Methods and controls for core welding support structures and reactor internals must conform to ASME Code, Section III, Subarticle NG-4000. The welds must be examined and meet the acceptance criteria specified in ASME Code, Section III, Subarticle NG-5000.

- Nondestructive Examination

  Nondestructive examination (NDE) shall conform to the requirements of ASME Code, Section III, Subarticle NG-2500. The acceptance criteria shall be in accordance with the requirements of ASME Code, Section III, Subarticle NG-5300.

- Austenitic Stainless Steels

SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," Subsections II.2 and II.4.a, b, d, and e, provide the acceptance criteria for the reactor internal materials.

RG 1.44 describes acceptance criteria for preventing intergranular corrosion of stainless steel components. Furnace-sensitized material should not be allowed, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steel from contamination during handling, storage, testing, and fabrication, as well as for determining the degree of sensitization that occurs during welding. RG 1.31 describes acceptable criteria for ensuring the integrity of welds in stainless steel components.

- Other Materials

  All materials used for reactor internals and core support structures must be selected for their compatibility with the reactor coolant, as described in ASME Code, Section III, Subarticles NG-2160 and NG-3120. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be specified to provide assurance that these materials will not deteriorate in service. Acceptable heat treatment temperatures are aging at 565 degrees C to 595 degrees C (1,050 degrees F to 1,100 degrees F) for type 17-4 PH and tempering at 565 degrees C (1,050 degrees F) for type 410 stainless steels. Other materials shall have similarly appropriate heat treatment and fabrication controls in accordance with strength and compatibility requirements.

4.5.2.1 Summary of Technical Information

DCD Tier 2, Section 4.5.2, describes the materials used to fabricate reactor internal and core support materials. Specifically, the DCD provides information about the materials specifications, controls on welding, NDE of wrought seamless tubular products, fabrication and processing of austenitic stainless steel components, and items concerned with materials other than austenitic stainless steel. Each of these topics is discussed below.

<u>Materials Specifications</u>

The DCD requires that all core support structures be fabricated from ASME-specified materials and designed in accordance with the criteria of ASME Code, Section III, and Subsection NG.

The other reactor internals are non-ASME Code, and they may be fabricated from ASTM or ASME specification materials or other equivalent specifications.

<u>Controls on Welding</u>

The DCD requires that core support structures be fabricated in accordance with the criteria of ASME Code, Section III, Subarticle NG-4000, and the examination and acceptance criteria included in Subarticle NG-5000. The reactor internals, other than the core support structures, meet the criteria of the industry standards (e.g., ASME or

American Welding Society), as applicable. The qualification criteria of ASME Code, Section IX, are followed in the fabrication of core support structures. All welds are made with controlled weld heat input.

Nondestructive Examination of Wrought Seamless Tubular Products

The DCD requires that the stainless steel CRD housings, which are partially core support structures (inside the reactor vessel), serve as the reactor coolant boundary outside the reactor vessel. The CRD housing material is supplied in accordance with the criteria of ASME Code, Section III Class 1. The CRD housings are examined and tested in accordance with ASME Code, Section III, Subsection NB for the pressure boundary portion of the housing and in accordance with ASME Code, Section III, Subsection NG for the non-pressure boundary portion.

The peripheral fuel supports are supplied in accordance with ASME Code, Section III Subsection NG. The material is procured and examined according to ASME Code, Section III Subarticle NG-2500.

Wrought seamless tubular products for other reactor internal components are supplied in accordance with the applicable ASTM or ASME materials specifications. These specifications require a examination on each length of tubing or pipe.

Regulatory Guide Conformance for the Fabrication and Processing of Austenitic Stainless Steel

The DCD requires that significantly cold-worked stainless steels not be used in the reactor internals except for vanes in the steam dryers. Applying limits on hardness bend radii, and surface finish on ground surfaces, controls cold work. Furnace-sensitized material is not allowed. Electroslag welding is not applied for structural welds. The delta ferrite content for weld materials used in welding austenitic stainless steel assemblies is verified on undiluted weld deposits for each heat or lot of filler metal and electrodes. The delta ferrite content is defined for weld materials as 5.0 ferrite number (FN) minimum, 8.0 FN average, and 20 FN maximum. This ferrite content is considered adequate to prevent any microfissuring (hot cracking) in austenitic stainless steel welds in compliance with RG 1.31.

The limitation placed on the delta ferrite in austenitic stainless steel castings is a minimum value of 8 percent and a maximum value of 20 percent. The maximum limit is used for those castings designed for a 60-year life, such as the fuel support pieces, to limit the effects of thermal aging degradation. Proper solution annealing of the 300-series austenitic stainless steel is verified by testing in accordance with ASTM A-262 “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels”. Welding of austenitic stainless steel parts is performed in accordance with ASME Code, Section IX (welding and brazing qualification), and Section II, Part C (welding rod electrode and filler metals).

All cleaning materials and process materials that contact stainless steel during manufacture and construction are controlled to prevent exposure to contaminants. Any inadvertent surface contamination is removed to avoid potential detrimental effects.

Special care is exercised to ensure removal of surface contaminants before any heating operation. Water quality for rinsing, flushing, and testing is controlled and monitored. The degree of cleanliness obtained by these procedures meets the criteria of RG 1.37.

Other Materials

The DCD specifies that hardenable martensitic stainless steel and precipitation hardening stainless steels are not to be used for the reactor internals. Materials, other than type 300 stainless steel, used in reactor internals are type or grade XM-19 stainless steel, niobium-modified Alloy 600 and N07750 (Alloy X-750), or equivalent. All niobium-modified Alloy 600 material is used in the solution-annealed condition and meets the criteria of ASME Code Case N-580-1. Alloy X-750 components are fabricated in the annealed and aged condition. In those areas that require maximum resistance to stress corrosion, the material is used in the high-temperature 1,093 degrees C (1,999.4 degrees F) annealed plus single aged condition.

Hard chromium plating surface is applied to austenitic stainless steel couplings. All materials used for reactor internals are selected for their compatibility with the reactor coolant as specified in ASME Code, Section III, Subarticle NG-3120.

4.5.2.2 *Evaluation*

The staff divided its evaluation of the reactor internals and core support materials in DCD Tier 2, Section 4.5.2, into five topics equivalent to those described in SRP Section 4.5.2. These topics include materials specifications, controls on welding, NDE, fabrication and processing of austenitic stainless steel components, and other materials and considerations.

4.5.2.2.1 Materials Specifications

DCD Tier 2, Table 4.5-1, does not identify many of the reactor internal components discussed in Section 2 of DCD Tier 1. In RAI 4.5-1, the staff requested the applicant to revise Table 4.5-1 in DCD Tier 2 to include all core support structures and reactor internal components used in the ESBWR with corresponding materials specifications. The staff also recommended that Table 4.5-1 in DCD Tier 2 be revised to differentiate the core support structure components from the reactor internal components. In response, the applicant proposed a revision to Table 4.5-1 of DCD Tier 2 to include all reactor internal and core support components discussed in Section 2 of DCD Tier 1. The applicant will replace DCD Tier 2, Table 4.5-1, in the next update of the DCD. The staff finds that this proposed revision contains sufficient information on the materials specifications for all significant reactor internal and core support structure components and, therefore, is acceptable. Based on the applicant’s response, RAI 4.5-1 was resolved.

DCD Tier 2, Section 4.5.2, contains no drawings of the core support structures or reactor internals. In RAI 4.5-2, the staff requested the applicant to provide the detailed drawings of all significant core support structures and reactor internal components, as well as assembly drawings to show how the core support structure components and reactor internal components are attached to each other and/or to the reactor vessel. The staff also suggested that DCD Tier 2, Section 4.5.2, include the drawings and diagrams. In response, the applicant provided reactor internals assembly drawings to supplement

Figure 2.1.1-1 in DCD Tier 1. The drawing shows the assembly of the major core structures and internal components listed in the revised Table 4.5-1. In addition, the applicant provided conceptual drawings of the shroud, top guide, chimney, chimney partition, and core plate-to-shroud joints. Based on the applicant's response, RAI 4.5-2 was resolved.

In its response to RAIs 4.5-18 through 4.5-20, the applicant also provided sketches of other reactor internal components. The applicant will include the assembly drawings in Section 3.9.5 of DCD Tier 2 when it is next updated. The staff finds that the drawings clarify how reactor internals and core support structures are assembled and supported. Therefore, this issue is closed. Based on the applicant's responses, RAIs 4.5-18, 4.5-19, and 4.5-20 were resolved.

DCD Table 4.5-1 identifies cast austenitic stainless steel as a material that will be used in the ESBWR reactor internals. The staff noted that cast austenitic stainless steel is susceptible to a loss of fracture toughness because of thermal aging embrittlement, neutron irradiation embrittlement, and void swelling in the reactor vessel. The staff's concern was based on a letter from Christopher I. Grimes of the NRC to Douglas J. Walters of the Nuclear Energy Institute, "License Renewal Issue No. 98-0030, 'Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,'". In addition, ultrasonic examinations of cast austenitic stainless steel have not been reliable. In RAI 4.5-3, the staff requested the applicant to address the aforementioned degradation mechanism and its concern about the ability to inspect components made with cast austenitic stainless steel.

The applicant responded that the use of cast stainless steel components for the ESBWR internals is very limited and confined to components that are common to previous BWR designs. As such, these components have more than 35 years of BWR operating experience with no known problems or failures. The one core support application is the fuel support casting. This is a removable, replaceable piece of hardware on which the fuel bundles sit. In this location, the casting is well below the bottom of active fuel length and, as such, sees relatively low neutron dose compared to other core support structures such as the shroud and top guide. Neutron-induced void swelling does not occur because both the temperature and fluence are well below the nominal thresholds for this phenomenon in stainless steels (see "Irradiation Temperature Dependence of Void Formation in Type 304 Stainless Steel," by Sandusky et al.). The applicant stated that thermal aging is also not a concern. At the normal operating temperature for all BWRs of 550 degrees F, thermal aging of low-carbon stainless steel castings with less than 20-percent ferrite is barely measurable (NUREG/CR-5385, "Initial Assessment of the Mechanisms and Significance of Low-Temperature Embrittlement of Cast Stainless Steels in LWR Systems," issued August 1990). To ensure that the potential for thermal aging is thoroughly limited, the applicant specified a maximum ferrite value of 20 percent for the castings.

The applicant stated further that the steam separator swirl generator castings and connector castings in the steam dryer are the only other castings in the ESBWR internals design. The swirler casting is a non-ASME Code, non-safety-related functional component that sees essentially no neutron dose because of its location. The only structural demands on this casting are from directing the flow of steam and water and supporting the minor weight of the individual separator assembly to which the casting is welded. As with the fuel support casting, this component is unchanged from early BWR designs, except that the carbon content is now limited to the L-grade range (i.e., low carbon), and ferrite content is controlled to a range of 8 to 20 percent. The steam dryer castings are also non-ASME Code, non-safety-related components that experience essentially no neutron dose. As with the castings mentioned above, they are low carbon with ferrite control. As such, the steam dryer castings are highly resistant to thermal aging and SCC. Because none of these castings is subject to ultrasonic testing, either in fabrication or in service, their ability to be inspected is not an issue.

The staff finds that, based on the applicant's response, thermal aging embrittlement, neutron irradiation embrittlement, and neutron void swelling are not a concern for the cast austenitic stainless steel in the ESBWR environment. In addition, the ability of the subject material to be inspected is moot because the steam dryer is a non-ASME Code, non-safety component that is not required to be inspected by ultrasonic testing. The staff finds that the applicant's use of cast austenitic stainless steel is acceptable; therefore, this issue is closed. Based on the applicant's response, RAI 4.5-3 was resolved.

DCD Table 4.5-1 identifies that non-L-grade 304 and 316 stainless steels are used for the reactor vessel internals and core support structures. In RAI 4.5-5, the staff requested the applicant to justify the use of non-L-grade 304 and 316 stainless steels in light of the industry experience of intergranular stress-corrosion cracking (IGSCC) in 304 and 316 stainless steel in the BWR environment. The applicant responded that the carbon content is limited and should not exceed 0.02 percent in all welded wrought austenitic stainless steel components in the ESBWR that are exposed to reactor water at temperatures exceeding 93 degrees C (199.4 degrees F). The difference between 304 and 304L (or 316 and 316L) is in their respective mechanical strengths. The applicant proposed to add a footnote to DCD Tier 2, Table 4.5-1, requiring that the carbon content of all type or grade 304/304L or 316/316L used in the core support structures and reactor internal components be limited to a maximum of 0.02 percent. The applicant will add this limitation to Table 4.5-1 in the next update of the DCD. The staff finds this explanation acceptable. Based on the applicant's response, RAI 4.5-5 was resolved.

The staff notes that IGSCC has occurred in 304 and 316 stainless steel material in the BWR coolant environment as a result of sensitization. The high carbon content in 304 and 316 stainless steel contributes to this sensitization. According to the materials specifications of SA-240 in the ASME Code, Section II, the carbon content for 304/316 and 304L/316L stainless steel is limited to 0.08 percent and 0.03 percent, respectively. However, as the applicant stated above, the DCD limits the carbon content of 304, 304L, 316, and 316L stainless steels to 0.02 percent, which is lower than the ASME Code specifications for either stainless steel type. The low-carbon-content requirement should minimize the potential for sensitization of 304 and 316 stainless steel, which in turn will minimize the potential for IGSCC. Therefore, the staff finds that the revised materials specifications for type 304 and 316 stainless steel in DCD Tier 2, Table 4.5-1, are acceptable.

In RAI 4.5-6, the staff requested the applicant to clarify the ASME Code and non-ASME Code components used in the reactor internals, identify the specific materials specification for each of the reactor internal components, and include this information in DCD Table 4.5-1. The applicant responded that those reactor internals with a core support function are fabricated and certified to ASME Code, Section III Subsection NG.

All other ESBWR internal components are considered “internal structures,” consistent with Subsection NG terminology. For these components, materials may be procured that meet either ASTM or ASME Code, Section II, standards or equivalents. The individual ASTM and corresponding ASME materials specifications are essentially identical (e.g., ASTM A-240, type 316L plate is identical to ASME SA-240, type 316L). The applicant proposed to revise DCD Table 4.5-1 in the next update to identify the materials specifications of reactor internals and core support structures. The staff finds that the proposed revision to DCD Table 4.5-1 provides specific ASME or ASTM materials specifications and clarifies the difference between ASME Code and non-ASME Code material. Therefore, based on the applicant’s response, RAI 4.5-6 was resolved.

In RAI 4.5-7, the staff requested the applicant to (1) discuss the operating experience (i.e., degradation) of the non-ASME Code materials used in the reactor internals in the current BWR fleet, (2) demonstrate that the non-ASME Code material will provide the strength, resistance to corrosion, and fracture toughness necessary to maintain the safe operation of the ESBWR, (3) discuss whether the non-ASME Code components are designed for and analyzed with the same loading combinations, in accordance with the ASME Code, Section III, as that used for the ASME Code components, and (4) clarify whether the non-ASME Code components are considered as safety or non-safety category components.

The applicant responded to RAI 4.5-7 with the following:

(1) As discussed in the response to RAI 4.5-6, the materials used for internal structures are identical in chemistry and properties to their ASME Code counterparts. Consequently, there is no distinction in behavior in BWR service between the ASME Code core support structures and other internal structures.

(2) Strength, corrosion resistance, and toughness of the internal structure materials are equivalent to that of their ASME Code counterparts.

(3) Internal structures are designed and analyzed using Article NG-3000 of ASME Section III, Subsection NG, as a guideline. Loading combinations are the same as those specified for core support structures. Stresses and fatigue usage factors will meet the limits specified in Subsection NG.

(4) Internal structures may be safety-related or non-safety-related, depending on their function. The standby liquid control line is an example of a safety-related internal structure. Non-safety internal structures include such components as the steam separators and steam dryer.

The staff determined that the applicant’s response is satisfactory. Based on the applicant’s response, RAI 4.5-7 was resolved.

In RAI 4.5-8, the staff requested the applicant to discuss which industry standards will be used for material selection, fabrication, construction, design, testing, and inspection for the non-ASME Code components. The applicant responded that the non-ASME Code materials used for the internal structures are identical in chemistry and properties to the ASME Code materials used for the core support structures. Consequently, there is no distinction in behavior in BWR service between the ASME Code core support structures and non-ASME Code reactor internal components. The applicant stated further that strength, corrosion resistance, and toughness of the non-ASME Code internal structure materials are equivalent to those of their ASME Code counterparts. Non-ASME Code internal structures are designed and analyzed using ASME Code, Section III Subsection NG-3000 as a guideline. Loading combinations are the same as those specified for core support structures. Stresses and fatigue usage factors will satisfy the limits specified in ASME Code, Section III, Subarticle NG. Internal structures may be safety-related or non-safety-related, depending on their function. The standby liquid control line is an example of a safety-related internal structure. Non-safety internal structures include components such as the steam separators and steam dryer. The applicant stated further that material selection and fabrication for the non-ASME Code components are consistent with the ASME Code. Welding procedures and welders are qualified to ASME Code, Section IX. Inspection methods are consistent with ASME Code, Section V, and the acceptance criteria that follow ASME Code, Section III Subsection NG.

The staff finds that the non-ASME Code components are designed and analyzed using ASME Code, Section III, as a guide. The structural performance of the non-ASME Code components in terms of strength, corrosion resistance, and toughness is equivalent to that of the ASME Code components. Therefore, based on the applicant's response, RAI 4.5-8 was resolved.

4.5.2.2.2 Controls on Welding

SRP Section 4.5.2 specifies that the methods and controls for core support structure and reactor internal welds must be performed in accordance with ASME Code, Section III, Division 1, Subarticle NG-4000, and the welds must be examined and meet acceptance criteria as specified in Subarticle NG-5000. However, DCD Tier 2, Section 4.5.2.2, "Controls on Welding," discusses the welding of the reactor internals without referring to the relevant ASME Code sections. In RAI 4.5-9, the staff asked the applicant to identify the ASME Code sections relevant to core support structure and reactor internal components that require welding and to describe the welding technique and procedures. In addition, the staff requested the applicant to clarify the intent of DCD Tier 2, Section 4.5.2.2, which does not explicitly mention welding.

The applicant responded that "fabrication," as used in DCD 4.5.2.2, is intended to encompass all fabrication processes, including welding as defined in ASME Code, Section III, Subarticle NCA-9000. For core support structures, the components are required to be built and certified in full compliance with ASME Code, Section III Subsection NG. Therefore, compliance with Subarticles NG-4000 and NG-5000 is implicit, and all welding will be performed and inspected accordingly. The applicant did not consider it necessary to explicitly refer to select portions of Subsection NG in the DCD because full compliance with Subsection NG in its entirety is required. For the non-ASME Code internal components, welding qualification according to ASME Code, Section IX, is required. Welding practices and inspections are generally consistent with ASME Code, Section III, NG-4000 and NG-5000. Most of the core support structures and reactor internals require some welding for assembly. The main exceptions are the fuel supports that rest on the core plate, which are machined from forgings or castings. Welding processes will be those commonly applied to stainless steels and nickel alloys, such as shielded metal arc welding, gas tungsten arc welding, submerged arc welding, and gas metal arc welding. Both manual and automatic processes will be applied. The specific welding techniques and procedures cannot be defined at this time because such details depend on the facility contracted to do the fabrication work.

The staff finds that the applicant has clarified the welding processes and referenced the relevant ASME Code sections for the core support structures and reactor internals. Based on the applicant's response, RAI 4.5-9 was resolved.

4.5.2.2.3 Nondestructive Examination

SRP Section 4.5.2.II.3, Draft Revision 3, issued April 1996, specifies that the acceptance criteria for NDE shall be in accordance with the requirements of ASME Code, Section III Subarticle NG-5300. However, DCD Tier 2, Section 4.5.2.3, does not specify the acceptance criteria for NDE. In RAI 4.5-10, the staff requested the applicant to include the acceptance criteria for NDE in DCD Tier 2, Section 4.5.2.3. The applicant responded that, for core support structures, full compliance with ASME Code, Section III, Subsection NG, is understood and so stated. Similarly, for the reactor internal components that have a pressure-retaining function, full compliance with ASME Code, Section III, Subsection NB, is required and so stated. These subsections contain acceptance criteria for NDE. Therefore, the applicant did not consider it essential for the DCD to explicitly reference individual articles of the ASME Code, such as NB/NG-5300. The staff finds that the applicant has clarified the ASME Code sections relevant to the acceptance criteria for NDE of the reactor internal and core support structure components. Based on the applicant's response, RAI 4.5-10 was resolved.

DCD Tier 2, Section 4.5.2.3 discusses the NDE of CRD housings and peripheral fuel supports but is silent on other reactor internal components. SRP Section 4.5.2.I.3, Draft Revision 3, issued April 1996, recommends that each product form in the reactor internals and core support structures be examined. In RAI 4.5-11, the staff requested the applicant to justify why NDE is not required for product forms other than CRD housings and peripheral fuel supports. The staff also requested that the applicant identify which specific tubular products will be hydrostatically tested. The applicant responded that it will revise DCD Tier 2, Section 4.5.2.3, to reflect the expanded scope of Draft Revision 3 to SRP Section 4.5.2.I.3. Examination of core support structure materials and welds will be in full compliance with ASME Code, Section III Subsection NG. In addition, pressure-retaining components and welds will be inspected in full compliance with Subsection NB. The applicant proposed to revise DCD Tier 2, Section 4.5, in the next update as described in the following sections.

Materials for core support structures will fully conform and be certified to ASME Code, Section III Subsection NG. Subarticle NG-2500 specifies the examination of materials (examination methods and acceptance criteria). Subarticle NG-5000 provides examination methods and acceptance criteria for core support structure weld edge preparations and welds. Tubular products that are pressure boundary components (CRD and in-core housings) will be examined according to ASME Code, Section III, Subarticle NB-2500, and associated pressure-retaining welds will be examined according to Subarticle NB-5000. For non-ASME Code reactor internal structures and associated welds, examinations are established based on relevant design and analysis information and will follow guidance from ASME Code, Section III, Subarticles NG-2500 and NG-5000, respectively.

The staff finds that the proposed revision to DCD Tier 2, Section 4.5.2.3, satisfies SRP Section 4.5.2, and, therefore, is acceptable. Based on the applicant's response, RAI 4.5-11 was resolved.

In RAI 4.5-14, the staff requested the applicant to discuss the pre-service inspection and inservice inspection program of all core support structure and reactor internal components. For each component, the staff requested that the discussion include specific examination techniques, frequency of the inspection, acceptance criteria, the area/coverage of the inspection, and the industry codes/requirements used. The staff also requested the applicant to provide a list of components that will not be inspected during the pre-service or in-service inspection activities and explain why the inspection is not needed.

The applicant responded that the pre-service and in-service inspections of core support structures and internal components are a COL holder issue. However, the applicant stated that visual examination of the core support structures will be performed during plant outages as required by ASME Code, Section XI Table IWB-2500-1, Item B13.40. The frequency of the examinations will be conducted as identified in Subarticle IWB-2400 of the ASME Code, Section XI. The examination personnel shall be qualified in accordance with Subarticle, IWA-2300. The ASME Code has no requirements for pre-service and in-service inspections of reactor internal components that are non-ASME Code components. These components include the chimney, chimney partitions and chimney restraints, chimney head and steam separator assembly, chimney head bolts, steam dryer assembly, feedwater spargers, standby liquid control piping and distribution headers, in-core guide tubes, in-core guide tube restraints, guide rods, and drain pipes.

The applicant stated further that during the fabrication of core support structures all material is examined as required by ASME Code, Section III, Subarticle NG-2500. For the examination of non-ASME Code internal components, the ASME Code is used as a guideline. A liquid penetrant examination is required on the weld preparation surfaces before welding and on all machined surfaces. The extent of NDE of welds is determined by the weld quality and fatigue factors (ASME Code, Section III Table NG-3352-1) applied to the weld joints in the design analysis. All welds, materials, and subassemblies not accessible for inspection in the completed assembly are inspected for quality and cleanliness before the last activity that results in their inaccessibility.

A visual examination of the completed components that meets the requirements of ASME Code, Section XI, Subarticle IWA-2210, performed in the shop, serves as a "preservice visual inspection." The same rigorous quality and cleanliness requirements are applied to the installation of the reactor internals in the field.

The staff finds that the general pre-service and in-service inspection of the reactor internal components and core support structures follows the ASME Code; therefore, based on the applicant's response, RAI 4.5-14 was resolved.

The BWR Vessel and Internals Project (BWRVIP) have published many guidelines related to the inspection of reactor internals. The NRC has approved some of the BWRVIP reports. In RAI 4.5-15, the staff requested the applicant to discuss which BWRVIP guidance and reports will be used. The applicant responded that the BWRVIP guidelines were written for maintenance, inspection, and repair of currently operating

BWRs and do not address new plant construction. Consequently, these guidelines are not specifically used to establish ESBWR requirements. However, ESBWR materials selection and controls are generally consistent with the EPRI "Advanced Light Water Reactor Utilities Requirements Document." The staff finds this explanation acceptable, and therefore, based on the applicant's response, RAI 4.5-15 was resolved.

4.5.2.3 Fabrication and Processing of Austenitic Stainless Steel Components

DCD Tier 2, Section 4.5.2.4, states that significantly cold-worked stainless steels are not used in the reactor internals except for vanes in the steam dryers. In RAI 4.5-12, the staff requested the applicant to justify the use of cold-worked materials in vanes, considering the adverse impact of the cold work on the microstructure of the material and the susceptibility of cold-worked materials to SCC. The applicant responded that some degree of cold working is necessary to form the steam dryer vane shape. This design is essentially unchanged from the earlier BWRs. Thus, over 35 years of operating experience with this design has accumulated, and no failures of vanes have been observed. The material has been updated to current low carbon standards, and maximum hardness is controlled to a level well below the threshold for SCC. Because the only function of the vanes is to direct steam flow, these parts experience virtually no sustained tensile stress.

The applicant stated further that, even if SCC were to occur, there is virtually no potential to create a loose part because the vane banks are contained between perforated plate assemblies. The staff finds this explanation to be acceptable. Therefore, based on the applicant's response, RAI 4.5-12 was resolved.

DCD Section 4.5.2.5 identifies Alloy X-750 as a material that will be used in the reactor internals. However, Alloy X-750 materials are susceptible to IGSCC because of equalized and aged heat treatment conditions (BWRVIP-41, "BWR Vessel and Internals Project: BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," EPRI 1012137). In RAI 4.5-13, the staff requested the applicant to (1) identify the reactor internal components fabricated from Alloy X-750, (2) provide information on the aging heat treatment process of Alloy X-750 (i.e., aging temperature and holding time), (3) justify how this aging process will help to minimize SCC, (4) provide the optimal hardness value that is required to minimize the susceptibility to SCC, and (5) discuss why Alloy X-750 is identified in DCD Tier 2, Section 4.5.2.5, but not in DCD Table 4.5-1.

The applicant responded that, other than the CRD components identified in DCD Tier 2, Section 4.5.1, the use of Alloy X-750 in ESBWR internal components is very limited. The only application positively identified at this time is a coil spring on the shroud head bolt (a non-safety-related component) and possibly a latch component in the steam dryer. Age hardening of Alloy X-750 will consist of a single-step treatment. The coil spring will be age hardened for 16 hours at 732 degrees C (1,350 degrees F). Any other shapes of Alloy X-750 will be age hardened at 704 degrees C (1,299.2 degrees F) for 20–21 hours. The annealing process performed before the aging treatment produces most of the improved SCC resistance of Alloy X-750. ESBWR Alloy X-750 components will be annealed at 1,080–1,108 degrees C (1,975–2,025 degrees F). This is the heat treatment condition developed for improved BWR jet pump beam performance in the early 1980s. It is consistent with the Type 3 heat treatment of the ASTM/ASME B/SB-637 material and the EPRI guidelines on Alloy X-750. The heat treatment, in conjunction with a single-step aging treatment, is considered to provide optimum stress-corrosion resistance to Alloy X-750 in BWR applications. Hardness has not been identified as a control parameter for SCC resistance; except that it is known that hardness in excess of Rockwell C40 (Rc40) can indicate elevated susceptibility. Type 3 heat treatment of ASTM/ASME B/SB-637 material specifies a Rockwell C40 maximum hardness. The applicant also committed to revising DCD Table 4.5-1 to include the known components fabricated with Alloy X-750. The staff finds the applicant's response acceptable because all Alloy X-750 components in the reactor internal components are non-safety-related, and any potential cracking of these components will not adversely impact safe operation of the reactor vessel. In addition, the Alloy X-750 components in the ESBWR will have undergone a high-temperature anneal treatment and a single-step aging treatment to improve their corrosion-resistance performance. Therefore, based on the applicant's response, RAI 4.5-13 was resolved.

In RAI 4.5-16, the staff asked the applicant to discuss the maintenance program for the bolts and threaded fasteners used in the core support structures and reactor internals to ensure their structural integrity and to prevent them from becoming loose parts in the reactor coolant system. The applicant responded that cracking of bolts and fasteners in the core support structure and reactor internal components has not been an issue in operating BWR plants. Positive locking mechanisms are used for bolting applications (e.g., nuts are tack welded in place to prevent them from coming loose), and visual inspections are performed during installation. Austenitic stainless steel bolts and nuts of types 304, 304L, 316, and 316L have been generally used in the past. However, in newer plants, including the ESBWR, nitrogen-strengthened austenitic stainless steel, grade XM-19, material is being used for high-load bolted joints. Because the ASME Code, Section XI, has no requirement for in-service inspection of the bolts and because of the favorable BWR operating experience, there are no formal ESBWR maintenance and inspection requirements for bolts and threaded fasteners inside the RPV. The staff finds this explanation acceptable. Based on the applicant's response, RAI 4.5-16 was resolved.

In RAI 4.5-17, the staff requested from applicant to identify the ASME Code requirements for material selection, inspection, design, fabrication, and construction of the chimney, chimney partitions, and chimney head; to describe the fabrication, assembly, and installation of the chimney, chimney partitions, and chimney head; and to discuss whether a mockup test of the chimney assembly in a reactor vessel environment has been conducted to verify the structural integrity of the chimney assembly.

The applicant responded that ASME Code, Section III, Subsection NG, was used as a guideline for the material, design, fabrication, and inspection of the chimney, chimney partitions, and chimney head. These components are classified as internal structures and do not require an ASME nuclear code stamp. As discussed above, they are non-ASME Code components. The chimney partition assembly consists of a grid of square structures, each of which encompasses 16 fuel assemblies and a bottom and a top ring.

The bottom ring rests on, and is pinned and bolted to, the bottom flange of the cylindrical chimney shell. The top ring of the assembly is supported against the inside of the chimney shell. The chimney assembly is bolted to the top guide and laterally supported by eight chimney restraints at the top. As discussed in Appendix 3L to DCD Tier 2, an air and water two-phase flow vibration test of both a 1/6-scale and a 1/12-scale model of a single chimney cell was performed. The results of the scale testing were extrapolated by a two-phase flow analysis to determine the characteristics of the pressure fluctuations acting on the partition wall of a full-size cell in steam-water conditions. The stress analysis showed an adequate margin against the allowable vibration peak stress amplitude based on the test results. The staff finds that, even though the chimney is a non-ASME Code component, ASME Code, Section III, Subsection NG, is used as a guide for the material selection, design, fabrication, and inspection of chimney components. Based on the applicant's response, RAI 4.5-17 was resolved.

The core shroud supports in the operating BWR fleet; are supported from and attached to the bottom of the reactor. However, for the ESBWR, the core shroud is attached and supported at the side wall of the reactor vessel, which may produce a bending moment on the vessel wall.

The staff was concerned that the shroud supports may not sustain the loads as calculated in the structural analysis because the vessel wall may not be as rigid as assumed in the analysis. In RAI 4.5-18, the staff requested the applicant to discuss whether the design of the core shroud supports considered the potential bending of the reactor vessel wall and whether the stress analysis of the reactor vessel shell considered the bending moment generated by the core shroud supports. In addition, the staff noted that the core shroud supports use niobium-modified Inconel 600 Alloy, which is susceptible to SCC. The staff requested the applicant to justify the selection of this material and to provide the drawings and design details, including the location and installation of the core shroud supports.

The applicant responded that shroud supports that are attached directly to the reactor vessel wall have been used in vessels built by Combustion Engineering (e.g., Plant Hatch). Analyses and experience have proven that the bending stresses produced by the cantilever shroud support design in these vessels are acceptable.

The bending moment from the shroud support will be included in the ESBWR design documentation containing the reactor vessel stress analysis. Since the bending moment from the ESBWR shroud support is smaller than in the aforementioned vessels because of a smaller gap between the shroud and the vessel wall, excessive bending stresses are not expected.

The applicant stated further that the core support material is Ni-Cr-Fe Alloy 600 with niobium added. ASME Code Case N-580-1 permits the use of niobium. Niobium-modified Ni-Cr-Fe Alloy 600 has been successfully used in the ABWRs, and tests have shown that it is highly resistant to SCC in a BWR environment. The staff finds it acceptable to include the bending moment of the core support structures in the reactor vessel shell analysis in the design documentation. In addition, the subject core shroud support design in currently operating BWRs has not adversely affected any reactor vessel walls. Therefore, the staff finds that the shroud support design is acceptable. Based on the applicant's response, RAI 4.5-18 was resolved.

In RAI 4.5-19, the staff requested the applicant to (1) provide assembly drawings of the CRD housing and stub tube to show how they are attached to each other and to the bottom of the vessel and (2) discuss weld joint details, welding processes, postweld heat treatments, materials to be used, and the fabrication sequence to be used to prevent sensitization of the stainless steel material (e.g., operating experience at Oyster Creek). In response, the applicant provided a schematic drawing of the reactor vessel CRD penetrations, which shows that the stub tubes are welded to the Ni-Cr-Fe cladding in the bottom head of the reactor vessel. The stub tube material is niobium-modified Ni-Cr-Fe Alloy 600 in accordance with ASME Code Case N-580-1. Welding of the joints between the stub tubes and the bottom head, and between the CRD housings and the stub tubes, is performed with a process using nickel Alloy 82 filler material, according to ASME SFA-5.14, Grade ER NiCr-3 (or use of Alloy 182; according to SFA-5.11, Grade ER NiCrFe-3 is not permitted). The final postweld heat treatment of the vessel is performed after the NiCrFe stub tubes are welded into the bottom head. This type of stub tube connection and material has been successfully used in the recent ABWRs. The staff notes; that the welds using nickel Alloy 82 filler material in pressurized-water reactors have experienced primary SCC. However, in the BWR environment, Alloy 82 weld metal is considered to be acceptable for use because of its resistance to IGSCC (NUREG-0313, Revision 2). In addition, the drawings provided by the applicant clarify how the CRD penetrations are attached to the stub tubes on that basis. Based on the applicant's response, RAI 4.5-19 was resolved.

Section 2.1.1 of DCD Tier 1 states that a lattice work of clamps, tie bars, and spacers provides lateral support and rigidity to the in-core guide tubes. In RAI 4.5-20, the staff requested the applicant to provide assembly drawings of the lateral support components and in-core guide tubes to show how the lateral support components are interconnected and how the in-core guide tubes are attached to the reactor vessel. The staff also requested that the applicant include the drawings in Section 4.5.2 of DCD Tier 2, identify materials used for the lateral support components and in-core guide tubes, and identify the number of penetrations.

The applicant responded that the lower ends of the in-core guide tubes are welded to the in-core housings in the bottom of the reactor vessel. The top ends extend through holes in the core plate, which provides lateral support. DCD Tier 2, Section 3.9.5, will include conceptual drawings illustrating the interconnections between the in-core guide tubes' lateral supports and their attachments to the lower portion of the shroud, and the connections between the guide tubes and the core plate. The revised Table 4.5-1, contained in the applicant's response to RAI 4.5-1, identified the material of the lateral supports and the in-core guide tubes. The reactor vessel bottom head has a total of 88 in-core penetrations. Figures 7.2-6 and 7.2-7 of DCD Tier 2 show the locations of the penetrations within the core. The applicant will revise DCD Tier 2, Section 3.9, to include this information in the next update. The staff finds that the applicant has provided drawings to clarify the configuration of the supports for the in-core guide tubes and associated lateral supports. Therefore, the staff finds that this issue is closed. Based on the applicant's response, RAI 4.5-20 was resolved.

DCD Tier 1, Section 2.1, states that special controls on material fabrication processes will be exercised when austenitic stainless steel is used for the construction of reactor internals. The staff issued In RAI 4.5-22, to request the applicant to describe the special controls that are used for material fabrication. The applicant responded that the controls will be contained in the detailed purchase specifications used to procure materials and fabricate components. Consequently, the full level of detail is not yet in place for the ESBWR. However, when these documents are prepared, the content will be very similar to existing specifications for ABWRs. For preparation of individual equipment documents, guidance will be taken from "Materials and Processes Controls," a top-level ESBWR materials and process document. The general practice is to have a materials specification that is used in conjunction with a fabrication specification for individual groups of equipment. For stainless steel materials, a number of controls are placed on the supplier that is more detailed than the basic ASTM and/or ASME requirements. In addition to the 0.02-percent maximum carbon limitation that will be included in the revised DCD Table 4.5-1, these equipment requirements documents will include the following among the controls generally applied:

- limitations on cobalt content (varies depending on proximity to the core);
- detailed controls on heat treatment time/temperature and quenching;
- for nuclear grade 304/316 material, confirmatory test of yield strength at 288 degrees C (550 degrees F);
- control of maximum hardness;
- sensitization test (modified ASTM A-262, Practice A);
- intergranular attack control;
- limitations/controls on weld repairs; and,
- cleaning, marking, and packaging controls

Fabrication of stainless steel components will be controlled using detailed fabrication specifications that include the following:

a. controls on hardness (e.g., control of mechanical cutting methods, machining controls, grinding controls, controls on cold bending, forming and straightening, and limitations on both bulk and surface final hardness);

b. controls on thermal processes (e.g., thermal cutting methods and heat input, hot forming and bending, and specific controls of induction bending);

c. welding controls (e.g., joint configurations, fit-up and gap, alignment, permitted processes, heat input control, back-purge and flux controls, allowed filler metals, ferrite control and measurement method, weld metal control and storage, and RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” restricted access qualification;

d. control of repairs, including allowed weld repairs;

e. detailed NDE requirements;

f. cleaning and cleanliness controls, including control of miscellaneous process materials; and,

g. traceability of material, marking, and packaging for shipment.

The staff finds that the applicant has satisfactorily identified the controls on the material fabrication processes, therefore, based on the applicant's response, RAI 4.5-22 was resolved.

4.5.2.4 *Other Materials and Considerations*

In RAI 4.5-21, the staff requested the applicant to clarify whether a hydrogen water chemistry program will be implemented in the ESBWR to mitigate SCC. The applicant responded that the materials were selected and process controls were identified without taking credit for the application of hydrogen water chemistry. The ESBWR design calls for the reactor internal components to be capable of operating for the design life of the ESBWR without experiencing SCC failures. The licensee may choose to adopt hydrogen water chemistry primarily for added corrosion resistance (no deleterious effects on the selected materials). The ESBWR design does incorporate features (e.g., injection taps) that facilitate installation of the hydrogen water chemistry system either before or after initial startup. The staff finds that it is appropriate for the ESBWR design to include features to facilitate future installation of the hydrogen water chemistry system. Based on the applicant's response, RAI 4.5-21 was resolved.

Operating BWRs have experienced cracking of the feedwater spargers (NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," November 1980). In RAI 4.5-23, the staff requested the applicant to describe design features, fabrication processes, and water chemistry to minimize or prevent cracking in feedwater nozzles and spargers in the ESBWR. The staff also requested the applicant to discuss the inspection program for the feedwater spargers.

The applicant responded that cracking of the feedwater spargers in some of the earlier BWRs was caused by three mechanisms: (1) high-cycle thermal fatigue caused by subcooled water leaking through the loose fit between the feedwater nozzle and the thermal sleeve, (2) subcooled water shedding from the subcooled thermal sleeve which periodically cooled the nozzle, and (3) thermal stratification in the feedwater sparger during low flow. In the ESBWR, the feedwater sparger, thermal sleeve, and vessel nozzle are welded together, thus eliminating the leakage flow of subcooled water. To prevent the reactor vessel nozzle from being exposed to cold water shedding from the thermal sleeve, licensees use a double thermal sleeve of a tuning fork design. The subcooled feedwater flows through the inner sleeve that is welded to the sparger. The concentric outer sleeve protects the vessel nozzle from being exposed to the cold water periodically shedding from the outer surface of the inner sleeve. The tuning fork design mitigates the thermal stresses between the austenitic stainless steel thermal sleeve and the low-alloy vessel nozzle. The ESBWR feedwater sparger has a row of spray nozzles mounted at the top of the sparger pipes so that the sparger will always be filled with water from the feedwater piping system, with minimal mixing with the warmer reactor vessel water. This sparger design helps to minimize thermal stratification within the sparger and piping during low flow conditions. Recent BWR product lines, as well as retrofit designs installed in the Monticello and Tsuruga-1 nuclear power plants in the early 1980s, have successfully used this sparger thermal sleeve design.

In reviewing the applicant's response to RAI 4.5-14, the staff finds that the applicant has adequately addressed the staff's concern about the potential for feedwater sparger cracking by specifying a design that has been used successfully in operating BWRs.

The staff finds that this issue is closed for the ESBWR DCD. Based on the applicant's response, RAI 4.5-23 has been resolved.

In RAI 4.5-24, the staff requested the applicant to describe the programs that will be used to prevent and manage metallic loose parts in the reactor vessel during fabrication/assembly, maintenance, normal operation, and refueling activities. The applicant responded that fabrication and installation of the reactor vessel and the reactor internals are performed in accordance with a quality program that meets the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. This includes implementation of a cleanliness program. Before plant operation, the vessel and attached piping will be flushed to remove debris that may have collected during construction. The licensee will implement loose part controls during service and maintenance. The staff is satisfied with the applicant's response because the DCD includes a quality program that conforms to the requirements of Appendix B to 10 CFR Part 50 to prevent loose parts. The staff finds that this issue is closed for the ESBWR DCD. Based on the applicant's response, RAI 4.5-24 was resolved.

In RAI 4.5-25, the staff requested the applicant to discuss the likelihood that the following degradation mechanisms will affect the core support structures and reactor internal components: ductile and brittle fractures, fatigue failures, distortion failures, wear failures, erosion (cavitation and liquid impingement), corrosion (pitting, leaching, galvanic, and intergranular), creep, corrosion fatigue, hydrogen-damage failures, embrittlement (neutron irradiation and thermal), and SCC. The staff also requested that the applicant discuss the steps taken to minimize or prevent these degradation mechanisms.

In a letter dated June 16, 2006, the applicant responded that the ESBWR is an evolutionary design that incorporates many aspects of prior BWR designs. In particular, the operating environment to which internal components are exposed is essentially identical. Therefore, more than 30 years of operating experience can be used to determine which degradation mechanisms may be active in the ESBWR. On that basis, the applicant addressed the aforementioned degradation mechanisms as follows:

- Ductile and brittle fractures: Use of ASME Code design rules ensure that there is no risk of ductile failures, even under upset conditions. Stainless steels and nickel alloys are not embrittled by fabrication processes, thermal aging, or exposure to BWR water. Although neutron irradiation decreases ductility, even at the highest exposure levels for reactor internals, significant residual toughness is retained.

- Fatigue failures: Fatigue failures have been very limited in BWR internals with a few exceptions. Historically, some fatigue failures have occurred in jet pump components, but jet pumps are not used in the ESBWR design. The other component that has experienced fatigue issues in operating BWRs is the steam dryer. The potential for fatigue failures in the ESBWR steam dryer is being addressed by implementation of a highly fatigue-resistant design based on extensive finite element and computerized fluid dynamic modeling, along with scale model testing.

- Distortion failures: Only one distortion failure has been observed in operating BWR internals. A series of steam dryers were fabricated with thin (0.125-inch) end hood plates, which became distorted by a pressure pulse generated by rapid MSIV closure. This problem was corrected by replacement with thicker hood material. The ESBWR steam dryer end hood plates are thicker than those used in some existing BWRs.

- Wear failures: Other than the CRDs and control blades, there are no moving parts in the ESBWR reactor internals. Wear has been considered by choosing hard-facing or wear-resistant alloys for moving components subject to wear. All the moving components that would potentially be subject to wear are routinely removable and replaceable.

- Erosion (e.g., cavitation and liquid impingement): This degradation phenomenon has not been observed in the internals of operating BWRs and is not expected in the ESBWR. Stainless steels are very resistant to erosion because of their high chromium content.

- Corrosion (e.g., pitting, leaching, galvanic, and intergranular): Stainless steels and nickel alloys have not been observed to experience corrosion phenomena in the BWR environment, which uses very pure deionized water.

- Creep: Stainless steels and nickel alloys do not experience creep at the maximum operating temperature of the ESBWR.

- Corrosion fatigue: A corrosion-fatigue interaction has not been observed in BWR internal components. The fatigue failures noted above are thought to have resulted from cyclic loading without any apparent or significant environmental factor. In any case, design improvements to eliminate the potential for fatigue failures in the ESBWR have addressed this concern.

- Hydrogen-damage failures: Hydrogen-driven failure mechanisms such as hydriding are not active in the BWR environment. Stainless steels and nickel alloys are not susceptible to hydrogen embrittlement or hydriding under the thermodynamic conditions in BWR water, even for a plant operating with hydrogen water chemistry.

- Embrittlement (neutron irradiation and thermal): Stainless steels and nickel alloys are not subject to thermal embrittlement at the ESBWR operating temperature (288 degrees C (550.4 degrees F)). Stainless steel does experience loss of ductility and toughness with neutron irradiation. This loss becomes significant at cumulative irradiation dose exceeding about 1×10^{21} n/cm^2 (E > than 1 million electron volts (MeV)). However, only certain areas of the reactor internals receive neutron doses exceeding this level, and even at the maximum dose for reactor internals, a significant degree of toughness is maintained. Operating BWRs achieve similar dose levels in reactor internals and no embrittlement failures have been observed, even in plants where there is frequent seismic activity.

- Stress-corrosion cracking: The design of the ESBWR addresses the potential for SCC of reactor internals by (1) use of only solution-annealed, low-carbon stainless steels and nickel alloys modified for high SCC resistance, (2) strict control of fabrication and installation processes, and (3) application of polishing to remove surface cold work in the weld heat-affected zones of the major structural welds in the large internals. These measures are expected to greatly reduce the potential for SCC of internals in the ESBWR relative to the currently operating BWRs. Routine inservice inspections will monitor the condition of the internals and be capable of detecting degradation by SCC in the unlikely event that it occurs.

The staff finds that the applicant has satisfactorily addressed the potential for degradation mechanisms by using appropriate material selection, fabrication, installation, and inspection of the core support structure and reactor internal components.

On the basis of the above evaluation, the staff concludes that the reactor internal and core support structure components of the ESBWR design satisfy the acceptance criteria of SRP Section 4.5.2. The ESBWR design also satisfies RG 1.31 for the control of ferrite content in stainless steel weld metal, RG 1.37 for the cleanliness and quality of the fluid system to minimize corrosion of the austenitic stainless steel and loose parts, RG 1.44 for the control of the use of sensitized stainless steel, and RG 1.84 for the use of NRC-approved ASME Code cases. Therefore, the ESBWR design satisfies the relevant requirements of 10 CFR 50.55a and GDC 1. Based on the applicant's response, RAI 4.5-25 was resolved.

4.5.2.5 Conclusions

On the basis of the information submitted, the staff concludes that the ESBWR design of the reactor internals and core support materials satisfies the relevant requirements of 10 CFR 50.55a and GDC 1 and, therefore, is acceptable. This conclusion is based on the fact that the ESBWR reactor vessel internals and core support structures satisfy ASME Code, Section III; RGs 1.31, 1.37, 1.44, and 1.84; and SRP Section 4.5.2.

4.6 <u>Control Rod Drive System</u>

The CRD system controls changes in core reactivity during power operation by movement and positioning of the neutron-absorbing control rods within the core in response to control signals from the RC&IS and rapid control rod insertion (scram) in response to manual or automatic signals from the RPS.

4.6.1 Regulatory Criteria

The staff reviewed DCD Tier 1, Section 2.2.2, "Control Rod Drive System," and Tier 2, Section 4.6, "Functional Design of Reactivity Control System," in accordance with SRP Section 4.6, Revision 2, "Functional Design of Control Rod Drive System." The staff's review covers the functional performance of the CRD system to confirm that the system can affect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents. Acceptance criteria are based on the following:

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

b. GDC 23, “Protection System Failure Modes,” as related to a failure of this system placing the reactor into a safe state

c. GDC 25, as related to the functional design of redundant reactivity systems to ensure that SAFDLs are not exceeded for a malfunction of any reactivity control system

d. GDC 26, "Reactivity Control System Redundancy and Capability", as related to the capability of the reactivity control system to regulate the rate of reactivity changes resulting from normal operations and AOOs

e. GDC 27, "Combined Reactivity Control Systems Capability" as related to the combined capability of reactivity control systems and the ECCS to reliably control reactivity changes to ensure the capability to cool the core under accident conditions

f. GDC 28, "Reactivity Limits" as related to postulated reactivity accidents

g. GDC 29, “Protection against Anticipated Operational Occurrences,” as related to functioning under AOOs

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

SRP Section 4.6 contains specific review procedures and acceptance criteria.

4.6.2 Summary of Technical Information

ESBWR DCD Tier 2, Section 4.6.1.2, describes the CRD system functions as follows:

a. controls change in-core reactivity by positioning neutron-absorbing control rods within the core in response to control signals from the RC&IS;

b. provides movement and positioning of control rods in increments to enable optimized power control and core power shape in response to control signals from the RC&IS;

c. provides the ability to position large groups of rods simultaneously in response to control signals from the RC&IS;

d. provides rapid control rod insertion (scram) in response to manual or automatic signals from the RPS so that no fuel damage results from any plant AOO;

e. in conjunction with the RC&IS, provides automatic electric motor-driven insertion of the control rods simultaneously with hydraulic scram initiation, which provides an additional, diverse means of fully inserting a control rod;

f. supplies rod status and rod position data for rod pattern control, performance monitoring, operator display, and scram time testing by the RC&IS;

g. in conjunction with the RC&IS, prevents undesirable rod pattern or rod motions by imposing rod motion blocks to protect the fuel;

h. in conjunction with the RC&IS, reduces the probability of a rod drop accident by detecting rod separation and imposing rod motion block;

i. in response to signals from the DPS, provides rapid control rod insertion (scram) and alternate rod insertion, an alternate means of actuating hydraulic scram, should an ATWS occur;

j. in conjunction with the RC&IS, provides for SCRRI and select rod input (SRI);

k. prevents rod ejection from the core as the result of a break in the drive mechanism, pressure boundary, or a failure of the attached scram line by means of a passive brake mechanism for the FMCRD motor, and a scram line inlet check valve;

l. supplies high-pressure makeup water to the reactor when the normal makeup supply system (feedwater) is unable to prevent the reactor water level from falling below the normal water-level range;

m. supplies purge water for the reactor water cleanup (RWCU)/shutdown cooling (SDC) system pumps; and,

n. provides a continuous flow of water to the nuclear boiler system to keep the reactor water-level reference leg instrument lines filled.

The CRD system consists of three major elements:

(1) electro-hydraulic FMCRD mechanisms;

(2) control rod drive hydraulic system (CRDHS); and

(3) HCU assemblies.

<u>Fine Motion Control Rod Drive</u>

The fine motion capability is achieved with a ball-nut and ball-screw arrangement driven by an electric motor. The ball-nut is keyed to the guide tube to prevent its rotation and traverses axially as the ball-screw rotates. 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. The electric motor-driven ball-nut and ball-screw assembly is capable of positioning the drive at a nominal 36.5-millimeter (1.44-inch) increments.

Control Rod Drive Hydraulic System

The CRD system provides electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods during abnormal operating conditions. High-pressure water stored in the individual HCUs provides the hydraulic power required for scram. Each HCU contains a nitrogen-water accumulator charged to high pressure and the necessary valves and components to scram two FMCRDs. Additionally, during normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs. The CRDHS 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 CRDHS is also the source of pressurized water for purging the RWCU/SDC system pumps and the nuclear boiler system reactor water-level reference leg instrument lines. Additionally, the CRDHS provides high-pressure makeup water to the reactor during events in which the feedwater system is unable to maintain reactor water level. This makeup water is supplied to the reactor via a bypass line off the CRD pump discharge header that connects to the feedwater inlet piping via the RWCU/SDC return piping.

Hydraulic Control Unit

Each HCU furnishes pressurized water for hydraulic scram, on signal from the RPS, to drive two CRD units. 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 nitrogen gas bottle provides a source of readily available high-pressure, high-discharge flow rate of nitrogen to the accumulator. The accumulator provides the stored energy necessary to obtain the required high-pressure, high-flow-rate discharge of water to the two associated FMCRDs. 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, check valves, and restricting orifice.

4.6.3 Staff Evaluation

ESBWR DCD Tier 2, Section 4.6.1.1.1, specifies the safety design bases of the CRD system as follows:

1. The design shall provide for the rapid control rod insertion (scram) so that no fuel damage results from any AOO.

2. The design shall include positioning devices, each of which individually supports and positions a control rod.

3. Each positioning device shall be capable of holding the control rod in position and preventing it from inadvertently withdrawing outward during any non-accident, accident, post-accident, or seismic condition.

4. Each positioning device shall be capable of detecting the separation of the control rod from the drive mechanism to prevent a rod drop accident.

5. Each positioning device shall provide a means to prevent or limit the rate of control rod ejection from the core resulting from a break in the drive mechanism pressure boundary. This is to prevent fuel damage resulting from rapid insertion of reactivity.

6. The design provides for isolation capability, which terminates high pressure make up water, HPCRD, to ensure containment pressure remains within design limits.

The staff's review of the functional design of the ESBWR CRD system confirmed that it satisfies the above safety design bases and the regulatory criteria in Section 4.6.1 of this report.

The staff's review of the functional design of the FMCRD system confirmed that the design has the following capabilities to satisfy the various reactivity control conditions for all modes of plant operations:

1. the capability to operate in full-power mode throughout plant life;

2. the capability to vary power level from full power to hot shutdown and have power distributions within acceptable limits at any power level; and,

3. the capability to shut down the reactor to mitigate the effects of postulated events, is discussed in Chapter 15 of this report.

The ESBWR design incorporates electric-hydraulic FMCRDs that will provide electric fine rod motion during normal operation and hydraulic pressure for scram insertion. A ball-nut and spindle arrangement driven by the electric motor provides fine motion during normal operation. In response to a scram signal, the control rods will be inserted hydraulically by the stored energy in the scram accumulator, similar to the means of insertion in the currently operating BWR CRDs.

A scram signal is also given simultaneously to insert the FMCRDs electrically via the FMCRD motor drive. This diversity, which includes both hydraulic and electric methods of scramming, provides a high degree of assurance of rod insertion on demand.

The FMCRD is designed to control reactivity during power operation. Automatic rod insertion will control reactivity in the event of fast transients.

If the reactor cannot be shut down with the control rods, the operator can actuate the SLCS (if not automatically started), which injects a solution of sodium pentaborate into the primary system. Section 9.3.5 of this report addresses the evaluation of the functional design of the SLCS.

Section 15.5 of this report discusses compliance with the ATWS rule, 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

Section 3.6 of this report discusses compliance with GDC 4 requirements that the CRDS be designed to perform its safety-related functions and not be compromised by adverse environmental conditions caused by high- or moderate-energy pipe breaks.

The FMCRD will control reactivity in the core by moving control rods interspersed throughout the core. These rods will control the reactor's overall power level and will provide the principal means of quickly and safely shutting down the reactor.

The staff issued RAI 4.6-7, to request the failure modes and effects analysis (FMEA) for the FMCRD. In their response the applicant proposed that the FMEA for the ABWR FMCRD system is applicable to the ESBWR because the ABWR and ESBWR FMCRD systems are similar (except for a few items). Appendix 15B to the ABWR DCD, Revision 4, provides the FMEA submitted for the ABWR. The text and descriptive material in Sections 15B.2.1 ("Introduction"), 15B.2.2 ("Conclusion"), 15B.2.3 ("Description"), and 15B.2.4 ("FMCRD Failure Modes Evaluation") are applicable to the ESBWR, with the exception that the FMCRD stepping motor of the ABWR design is replaced with the induction motor/magnetically coupled FMCRD design of the ESBWR. The staff reviewed the submitted material and concluded that the ABWR FMEA are unchanged by this difference and so are appropriate and applicable for the ESBWR FMEA analysis. The staff accepts this response, therefore, based on the applicant's response, RAI 4.6-7 was resolved.

The single-failure analysis of the FMCRD and HCU components indicates that the system design is satisfactory. A supply pump (with a spare pump on standby) will provide the HCUs with water from the condensate and feedwater system or the condensate storage tank to supply CRD purge water and to supply the purge water to the RWCU pumps. The supply pump also will provide water to a scram accumulator in each HCU to maintain the desired water inventory. When necessary, the accumulator will force water into the drive system to scram the control rods connected to that HCU; the volume of water in the scram accumulator will be sufficient to scram two rods. A single failure in an HCU may result in the failure of two control rods. Section 4.3 of this report discusses the impact of this feature on shutdown margin.

The FMCRD is designed to permit periodic functional testing during power operation with the capability to independently test individual scram channels and the motion of individual control rods. The FMCRD is designed so that failure of all electrical power or instrument air will cause the control rods to scram, thereby protecting the reactor. This feature meets the requirements of GDC 23.

Preoperational tests of the CRDHS will be conducted to verify the capability of the system. Startup tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week.

After each refueling shutdown, control rods will be tested for compliance with scram time criteria from the fully withdrawn position. Section 14.2 of this report presents the staff's evaluation of the preoperational and startup tests.

The FMCRD is designed to control reactivity under normal operating conditions and during AOOs and infrequent events. The safety analyses discussed in DCD Chapter 15 demonstrate this capability. The CRD system also will be capable of holding the core subcritical under cold shutdown conditions. The SLCS will be capable of bringing the reactor subcritical under cold-down conditions if the control rods cannot be inserted. These protection and reactivity control systems, taken together, satisfy the requirements of GDC 26, 27, and 29 pertaining to reactivity control system redundancy and capability, combined reactivity control system capability, and protection against AOOs and infrequent events.

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns will be selected to achieve optimum core performance and low individual rod worth. The RC&IS will reduce the chances of withdrawal other than by the preselected rod withdrawal pattern. The reactor plant control system function will assist the operator with an effective backup control rod monitoring routine that enforces adherence to established control rod procedures for startup, shutdown, and low-power-level operations. A malfunction in these systems could result in either a local or global reactivity change. Chapter 15 of this report includes analysis of accident scenarios such as control rod withdrawal error. As part of that review, the staff evaluated the categorization of these reactivity events, their acceptance criteria, and compliance with GDC 25, as discussed in Section 4.2 and Chapter 15 of this report, specifically regarding RAI 4.2-6. The staff reviewed the compliance of the CRD system with GDC 25. Based on the applicant's response, RAI 4.2-6 was resolved in Chapter 15.

The safety concerns associated with a pipe break, described in NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," issued August 1981, are not applicable to the ESBWR. The ESBWR design does not include scram discharge volume piping. The water displaced by the CRD during the scram will be routed to the RPV.

The high-pressure makeup mode of operation initiates automatically on receipt of a low-water level 2 signal. The CRDHS supplies high-pressure makeup water to the reactor vessel (about 1,000 gallons per minute with both pumps running simultaneously) through the RWCU/SDC. The flow is then routed through the feedwater system sparger for delivery to the reactor. At high reactor water level 8, the high-pressure makeup flow control valve closes to stop flow to the reactor to prevent flooding of the main steamlines. Since the ESBWR has no high-pressure core makeup system, the high-pressure core makeup mode of operation is an important feature of the FMCRD system.

The CRD pumps are tripped by coincident low-water levels in two of the three GDCS pools to prevent containment pressurization.

In RAI 4.6-10, the staff requested the applicant to identify the portions of the CRD system that are safety-related and to describe how the safety-related portions of the system are isolated from the nonessential portion of the system.

In response, Applicant identified the following safety-related CRD system equipment:

4. FMCRDs, including the following parts:

   – primary pressure components;

   – hollow piston;

- labyrinth seal;
- latches;
- guide tube;
- brake (passive holding function);
- check valve;
- check valve retainers;
- internal anti-shootout (includes outer tube, outer tube to middle flange weld, and middle flange);
- parts that couple the brake with the hollow piston;
- separation switches; and,
- antirotation device.

5. HCU (scram circuit only);
6. scram insert piping;
7. scram charging header pressure instrumentation; and,
8. high-pressure makeup piping at the connection to the RWCU/SDC system (including the check valve and injection valve).

According to the applicant, the CRD System is arranged in a manner that separates the safety-related equipment from the non-safety-related portions of the system. The FMCRDs are mounted to the reactor vessel bottom head inside the primary containment. The HCUs are housed in four dedicated rooms located directly outside of the primary containment at the basemat elevation of the reactor building. These rooms are arranged around the periphery of the primary containment wall. Each HCU room serves the FMCRD associated with one quadrant of the reactor core. The HCUs are connected to the FMCRDs by the scram insert piping that penetrates the primary containment wall.

The balance of the non-safety-related hydraulic system equipment (pumps, valves, filters, etc.) is physically separated from the HCUs and housed at a different elevation in the reactor building. It is connected to the HCUs by three non-safety-related piping headers: the FMCRD purge water header, HCU charging water header and scram air header. As shown in DCD Figure 4.6-8, these headers are classified as seismic Category II so that they will maintain structural integrity during a seismic event and not degrade the functioning of the HCUs.

The high pressure makeup piping at the connection to RWCU/SDC is classified as safety-related seismic Category I piping to provide interface compatibility with the safety-related seismic Category I piping of the RWCU/SDC.

As described above, the safety systems are adequately separated from the non-safety system and hence RAI 4.6-10 was resolved.

ESBWR DCD Tier 2, Section 4.6.2.1.3, describes design features aimed at precluding excess reactivity events (e.g., rod ejection and rod drop events). The control rod mechanical design incorporates a brake system and ball check valve, which reduces the chances of rapid rod ejection. The ball check valve is classified as safety-related because it actuates to close the scram inlet port by reverse flow under system pressure, fluid flow, and temperature conditions caused by a break of the scram line. This prevents the loss of pressure to the underside of the hollow piston and the generation of loads on the drive that could cause a rod ejection. This engineered safeguard will protect against a high-reactivity insertion rate from a potential control rod ejection. Normal rod movement and the rod withdrawal rate will be limited through the FMCRD. Applicant adopted an internal CRD housing support to replace the support structure of beams, hanger rods, grids, and support bars used in current BWR designs.

This system will use the outer tube of the drive to provide support. This tube will be welded to the drive middle flange and will be attached by a bayonet lock to the guide tube base. The guide tube, supported by the housing extension, will prevent downward movement of the drive in the event of housing failure. The CRD housing support is designed to prevent ejection of a CRD and attached control rod.

The FMCRD is designed to detect separation of the control rod from the drive mechanism. Two redundant and separate safety related switches will be provided to detect the separation of either the control rod from the hollow piston or the hollow piston from the ball-nut. Actuation of either of these switches will cause an immediate rod block and will initiate an alarm in the control room, thereby reducing the chances of a rod drop accident. Because of the design features described, the ESBWR control rod design does not include a velocity limiter.

Based on these design features, the applicant believes that the ESBWR design incorporates sufficient safeguards to negate its susceptibility to excess reactivity events. Initially, the ESBWR DCD did not include design requirements or a CRDA analysis. The staff had concerns that several scenarios might lead to an excess reactivity event and that each case would require scenario specific analysis to ensure that it was beyond design basis. If any scenario was credible, acceptance criteria (e.g., coolability, radiological consequences) would need to be developed and an acceptable accident analysis performed to demonstrate that these criteria were satisfied. The inclusion of this family of accidents may involve changes to the proposed ESBWR TSs (e.g., limiting conditions for operation (LCOs), engineered safety features actuation system (ESFAS) setpoints) and the ESBWR DCD (e.g., Sections 4.2 and 4.6 and Chapter 15).
RAI 4.6-23 requested further information on the ESBWR design features and the probability and consequences of each accident scenario leading to an excess reactivity event. The staff reviewed the control rod drop event frequency estimates provided by applicant in response to RAI 4.6-23. The design and testing of the control rod and CRD mechanism include a number of diverse and redundant features for preventing a rod drop event, which is an indicator of high reliability in the design. Based on its review of

key design and operational features and the fault-tree analysis provided by the applicant, the staff concluded that the applicant has presented a reasonable estimate of the rod drop frequency. However, the staff has also considered the applicant's control rod drop event frequency evaluation provided in response to RAI 4.6-23, S01. Based on the potential consequences of an unrestricted reactivity excursion and to ensure compliance with GDC 28, the staff concluded that the ESBWR design must demonstrate RCPB integrity and acceptable radiological consequences for the CRDA irrespective of the probability of CRDA. SRP Section 4.2, Appendix B, offers more detailed regulatory criteria and guidance. The staff required this regulatory position to be updated in ESBWR DCD Tier 1, Section 2.2.2; Tier 2, Section 4.6; and Tier 2, Section 15.4.6. In its subsequent response to this RAI, Applicant met the SRP Section 4.2, Appendix B, criteria by analyzing the CRDA. Section 15.4.6 of this report includes the staff evaluation of the CRDA. Based on the results of the staff evaluation of Section 15.4.6 and the applicant's response, RAI 4.6-23 was resolved.

ESBWR DCD Tier 2, Section 4.6.1.2.2, describes the support of the fuel assemblies and the core support plate. DCD Tier 2, Section 4.1.2.1.2 states, "Each guide tube, with its orificed fuel support, bears the weight of four assemblies and is supported on a CRD penetration nozzle in the bottom head of the reactor vessel" The staff issued RAI 4.6-26, to request additional information concerning the design margin between the control rod guide tube flange elevation and core support plate elevation. Specifically, the staff requested that the applicant address: (1) thermal expansion and contraction of the reactor vessel; and (2) differential growth between the reactor vessel and the control rod guide tube.

In its response, the applicant stated that there is no contradiction between DCD Tier 2, Subsections 4.6.1.2.2 and 4.1.2.1.2. Subsection 4.1.2.1.2 describes the reactor configuration in its normal state. In this condition the weld between the CRD housing and the CRD penetration nozzle in the reactor bottom head carries the full weight of the four assemblies, the orificed fuel support, the control rod guide tube and the FMCRD. Subsection 4.6.1.2.2 describes the rod ejection condition in which the weld between the CRD housing and CRD penetration nozzle fails completely. In this case the control rod guide tube drops down a distance equal to the normal gap until its flange at the top engages with the core plate. Based on the above response the staff considers RAI 4.6-26 was resolved.

ESBWR DCD Tier 2, Section 4.6.1.2, describes the CRD system functions, including the "ability to position large groups of rods simultaneously." With the ability to move multiple control rods simultaneously comes the possibility to inadvertently move multiple rods. This inadvertent withdrawal would introduce a corewide power transient that would be more global than the traditional localized rod withdrawal error event. The inclusion of this accident may involve changes to the proposed ESBWR TSs (e.g., LCOs, ESFAS setpoints) and the ESBWR DCD (e.g., Sections 4.2 and 4.6 and Chapter 15). The staff issued RAI 4.6-27 to request more information on the core and plant systems' response to a rod withdrawal error event involving large groups of rods. The applicant responded that mitigation of spurious rod movement by one or more rods is provided by RC&IS functions. A rod withdrawal error at power is protected by the RWM and automated thermal limit monitor subsystems of the RC&IS that terminate any spurious rod movement of one or more rods before violation of the OLMCPR. Any disagreement between the two RC&IS channels initiates a rod block (unless one is bypassed). Any one channel can signal rod block. Section 15.3 of this report addresses rod withdrawal error during power. Based on the applicant's response and the results of the review of Section 15.3, the staff considers RAI 4.6-27 was resolved.

ESBWR DCD Tier 2, Section 4.6.1.2, describes the CRD system functions, including the provision of SCRRI. An inadvertent control rod run-in would result in a redistribution of core power and potentially an approach to a fuel design limit. The inclusion of this accident may involve changes to the proposed ESBWR TSs (e.g., LCOs, ESFAS setpoints) and the ESBWR DCD (e.g., Sections 4.2 and 4.6 and Chapter 15). The staff issued RAI 4.6-28 to request additional information on the core and plant systems' response to an inadvertent control rod run-in event. The applicant responded by stating that SCRRI is an automatic function of the RC&IS and CRD system in the ESBWR design. The CRD system also provides FMCRD run-in. This automatic ATWS mitigation feature uses the FMCRDs to run in all the control rods in an emergency. The applicant enhanced the SCRRI function in DCD Tier 2, Revision 3, to include simultaneous hydraulic insertion of rods, known as SRI. (See DCD Tier 2, Section 7.1.5.4.10.) With the addition of SRI, an inadvertent SCRRI/SRI actuation as described below does not challenge core thermal limits. The quick response of the SRI rods reduces core power without creating an axial power transient that could potentially challenge fuel thermal limits. DCD Figure 15.2-4 shows the response to a generator load rejection with turbine bypass. Except for the slight pressure transient at the beginning of the event, the response is very similar to an inadvertent SCRRI/SRI. As shown, the SRI quickly reduces the core power. Although the radial power distribution does change, the core power reduction is significant enough to ensure that thermal limits are not challenged. Analysis shows that an inadvertent run-in of a single FMCRD would not challenge thermal limits. In a follow-up the staff issued RAI 4.6-28 S01 to request additional information regarding the instances of SCRRI and/or SRI failure that may affect core symmetry in power distribution. In its response applicant satisfactorily addressed functions of SRI and SCRRI, partial SCRRI failure, and partial SRI insertion disturbing core symmetry and introducing instabilities. Staff review of this issue is found in Section 15.3 of this report, based on the applicant's response, RAI 4.6-28 S01 is considered resolved.

ESBWR DCD Tier 2, Section 4.6.1.2.1, describes the spring-loaded latches on the hollow piston that engages slots in the guide tube. These latches support the control rod and hollow piston in the inserted position following a scram. The staff issued RAI 4.6-29 to request additional information regarding possible latch failure and the significant power peaking and loss of shutdown margin. In their response the applicant stated that the holding function of these latches will be tested and confirmed via the continuous full-in position indicator light as part of the scram testing defined in ITAAC 12 in DCD Tier 1, Table 2.2.2-7. The applicant also included details of the slot locations in the guide tube wall. Based on the applicant's response and the defined testing in ITAAC 12 the staff considers RAI 4.6-29 as resolved.

ESBWR DCD Tier 2, Section 4.6.1.2.2, states, "Each FMCRD provides two position detectors, one for each control system channel, in the form of signal detectors directly coupled to the motor shaft through gearing." This section goes on to state, "This configuration provides continuous detection of rod position during normal operation." The staff issued RAI 4.6-30 to request additional information regarding the accuracy of the position indication. In their response to RAI 4.6-30 regarding the accuracy of the control rod position indicators, the applicant stated that the signal detectors sense the number of rotations of the FMCRD ball screw and translate that information into an analog signal corresponding to control rod position. The cited position accuracy comprises the variation in braking distance and the accuracy of position detection. The applicant stated that this system configuration is identical to that in the ABWR design and is based on European FMCRD designs that have many years of reliable operating experience. Based on the applicant's response regarding the role of the ball screw and that it has been applied to the ABWR the staff considers RAI 4.6-30 as resolved.

ESBWR DCD Tier 2, Section 4.6.1.2.2, describes the FMCRD components. This section discusses the spring-loaded control rod separation mechanism. The staff issued RAI 4.6-31 to request additional information regarding its concerns that over time, irradiation-induced spring relaxation might impact the ability of this mechanism to perform its safety-related function. In its response the applicant stated that these mechanisms would not be exposed to any significant neutron fluence because of the shielding provided by several meters of water in the reactor vessel between the core plate and the vessel bottom head. The staff agrees with this explanation. Based on the fact that there are several meters of water between the bottom of active core and the bottom of the vessel, and the significant neutron attenuation RAI 4.6-31 as resolved.

ESBWR DCD Tier 2, Section 4.6.1.2.2 also discusses the FMCRD electromechanical brake and states that a "braking torque of 49 N-m (minimum) and the magnetic coupling torque between the motor and the drive shaft are sufficient to prevent control rod ejection in the event of failure in the pressure retaining parts of the drive mechanism." The staff issued RAI 4.6-32 to request calculational information of the minimum torque required to prevent rod injection. In its response the applicant provided details of this calculation which, when based on conservative inputs, shows that the calculated torque on the ball screw (resulting from loading associated with a break in the scram line) remains below the 49 Newton-meter) (N-m) design breaking torque requirement. The staff finds this calculation acceptable. The minimum holding torque of 49 N-m will be verified as part of ITAAC 15 in DCD Tier 1, Table 2.2.2-7. Based on the applicant's response and the minimum value of 49 N-m, RAI 4.6-32 is considered resolved.

ESBWR DCD Tier 2, Section 4.6.3.5, states, "A test of the scram times at each refueling outage is sufficient to identify any significant lengthening of the scram times." ESBWR TS Surveillance Requirement 3.1.4.2 requires routine (e.g., every 200 days) sampling of scram times for a representative set of control rods. Based on recent experience with channel bow, the staff believes that routine scram tests are necessary to detect the onset of control blade interference resulting from channel bow and to ensure control rod operability and satisfaction of scram time requirements. The staff issued RAI 4.6-33 to request additional information regarding the planned testing to detect control blade interference. In its response the applicant stated that it did not intend to remove the routine scram testing. A subsequent revision to this DCD section clarified the requirement for routine testing. Based on the applicant's response and the revision of the DCD regarding testing RAI 4.6-33 is considered resolved.

DCD Tier 2, Section 4.6.1.2.6, describes a rod withdrawal block signal generated because of rod-gang misalignment. The staff issued RAI 4.6-34 to request additional information on the allowable gang misalignment (before rod block), the accuracy of measuring the misalignment, and whether any safety analysis or LCO accounts for this misalignment. The applicant's response included the explanation that the rod action and position indication A and B monitor the gang rod position and issue a rod block by sending appropriate rod block signals to the logic of the rod server processing channels in the remote communication cabinets if the gang misalignment exceeds a predetermined value. Section 15.3 of this report discusses rod gang misalignment in more detail. The staff reviewed the supplemental information and based on the applicant's response, RAI 4.6-34 was resolved.

ESBWR DCD Tier 2, Section 4.6.3.5, describes the surveillance test for the high-pressure makeup mode but does not state the frequency for this surveillance. The staff issued RAI 4.6-35 to request the frequency of the surveillance reported in Section 4.6.3.5. In their response the applicant stated that it intended a test frequency is comparable to that for a safety-related, motor-driven, high-pressure ECCS pump. A subsequent revision to this DCD section included the surveillance tests and frequencies; therefore, based on the applicant's DCD revision RAI 4.6-35 was resolved.

Standard TSs requires certain surveillance tests following maintenance and before declaring a system operable. DCD Tier 2, Section 4.6.3.5, includes no such requirements. The staff issued RAI 4.6-36 to request DCD modifications that reflect post-surveillance testing. In their response the applicant stated that scram time tests were required on each affected control rod following maintenance. A subsequent revision to this DCD section reflected this requirement, therefore; based on the proposed DCD modification RAI 4.6-36 was resolved.

The ESBWR CRD system design represents a departure from that of the currently operating BWR fleet. The staff issued RAI 4.6-37 to request discussion of the CRD operating experience in systems similar to ESBWR. The staff noted that the proposed CRD differs significantly from the US operating fleet CRDs. In their response regarding reactor operating experience with similar CRD system designs, the applicant described the commercial service of a similar design in the Japanese ABWRs. In approximately 20 reactor-years of service, these reactors have experienced no anomaly indicating a fundamental or serious design issue. Furthermore, the regulatory oversight agency has received no reports of related safety incidents. With respect to manufacturing and testing experience, FMCRDs and HCUs have been manufactured to design specification both for the Japanese ABWRs and for an ongoing Taiwanese project and have successfully passed performance testing requirements. The operating and manufacturing experience supplied by the applicant provides reasonable assurance that the ESBWR CRD system can be manufactured to satisfy all design requirements. Based on the applicant's response regarding ABWR operating experience, RAI 4.6-37 was resolved. The Tier 1 ITAAC will ensure that the CRD system installed at each ESBWR site satisfies these requirements.

Section 7.8.3 of this report includes the staff evaluation of the requirements pertaining to 10 CFR 50.62, the alternate rod injection system, and redundant scram air header exhaust valves.

4.6.4 Conclusions

The staff concludes that the functional design of the reactivity control system conforms to the requirements of GDC 4, 23, 25, 26, 27, 28, and 29 and 10 CFR 50.62(c)(3) (as it relates to the alternate rod injection system and redundant scram air header exhaust valves scram capabilities) with regard to demonstrating the ability to reliably control reactivity changes under normal operation, AOOs, infrequent events, and accident conditions including single failures. The design of the reactivity control system conforms to the applicable acceptance criteria of SRP Section 4.6 and hence is acceptable.

4. A ESBWR Stability

The staff focused its review of ESBWR DCD Tier 2, Section 4D, "Stability Evaluation," on the acceptability of the ESBWR design in meeting regulatory requirements. The staff reviewed the applicant's methodology for calculating stability margins during the pre-application phase. In the SER for NEDE-33083P, the staff accepted the TRACG4 code and the applicant's associated methodology for calculating ESBWR stability margins.

Since the staff had previously reviewed the method for determining stability margins in detail, this evaluation focuses on a review of the ESBWR as it relates to meeting regulatory criteria for stability and stability during ATWS. Section 4A.1 below documents the staff's review of ESBWR stability and the basis for meeting regulatory criteria. Section 4A.2 documents the staff's review of ESBWR stability during an ATWS event and the basis for meeting regulatory criteria.

4.A.1 ESBWR Stability

4.A.1.1 Regulatory Criteria

The staff reviewed ESBWR stability (ESBWR, DCD Tier 2, Section 4D) based on the guidance in SRP Section 15.9, "Boiling Water Reactor Stability," which lists the following high-level requirements for BWR stability reviews:

1. GDC 10, "Reactor Design," requiring the reactor design (reactor core, reactor coolant system, control and protection systems) are designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOO;

2. GDC 12, "Suppression of Reactor Power Oscillations," requiring that power oscillations that can result in conditions exceeding SAFDLs are not possible, or can be reliably and readily detected and suppressed.

3. GDC 13, "Instrumentation and Control," requiring a control and monitoring system to monitor variables and systems to assure adequate safety including those that can affect the fission process over their anticipated ranges for normal operation, AOOs, and accident conditions.

4. GDC 20, "Protection System Functions," requiring, in part, a protection system that automatically initiates the operation of the appropriate systems including the reactivity control systems, to assure that fuel design limits are not exceeded as a result of AOOs.

5. GDC 29, “Protection against Anticipated Operational Occurrences,” requiring, the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

6. GL 94-02, “Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors,” dated July 11, 1994, requires all reactors to install a stability LTS to satisfy GDC 10 and 12.

7. SRP Section 15.9, “Boiling Water Reactor Stability,” issued March 2007, defines specific acceptance criteria to ensure that the high-level requirements are satisfied for BWR stability reviews. Specific criteria applicable to the ESBWR design are described below.

To meet the requirements of GDC 12, the reactor core and its systems should be designed with sufficient margin to be free of undamped oscillations and other thermal-hydraulic instabilities for all conditions of steady-state operation (including single-loop operation and extended cycle operation with reduced feedwater temperature where these operating conditions are proposed) and for AOOs. The design should consider the following:

- If potential oscillations cannot be eliminated, design proposals must detect and suppress (D&S) them reliably and readily.

- A reactor is considered stable if it satisfies one of the following criteria:

    A. The calculated decay ratio (DR) for all three common stability modes (corewide, regional, and channel) satisfies the relationship DR less than (1 - σ) where σ is the uncertainty of the calculation. Staff must review and approve both the calculation methodology and its uncertainty. The value of σ is typically 0.2 but is methodology dependent. This value includes the code uncertainty and some degree of variability of the input parameters.

    B. Use of an approved correlation to estimate the regional stability mode based on calculated corewide and channel DRs is permitted. One example is the FABLE/BYPSS stability criteria reviewed and approved by the NRC staff and documented in NEDO-31960 “Long-Tem Stability Solutions Licensing Methodology,” issued May 1991.

- The staff has reviewed and approved a number of stability LTSs. As reactor and fuel designs evolve, the industry may propose new stability LTSs. The following criteria judge the acceptability of new stability LTSs and facilitate meeting the requirements of GDC 20:

    A. The LTS must protect against SAFDL violations automatically.

    B. The LTS must demonstrate by analysis that either (i) the probability of instabilities in the allowed operating region is sufficiently small or (ii) unstable power oscillations can be detected and suppressed readily without SAFDL violations. The LTS may use a combination of both demonstrations for different instability modes.

C. If the licensing-basis option is declared inoperable, the LTS must provide a backup option, which may implement manual or administrative actions only if operator actions required to prevent SAFDLs can be accomplished within the 2 minutes allowed for operator action in the demonstration calculations.

D. The LTS option must include generic TSs that address:

(i) the methodology for setpoint and region calculation and documentation of the setpoint on a cycle-specific basis (e.g., COLR)

(iii) operability and surveillance requirements for the licensing-basis option

(iv) a time limit (120 days maximum) for operation under the backup option

- To meet requirements of GDC 13, stability-related instrumentation functionality must be demonstrated by analysis. Hardware implementation must follow SRP Section 7.2 requirements.

- In addition to the density wave instability modes, the applicant must ensure that the plant is free from other instability modes that could violate SAFDLs (e.g., startup or control system instabilities) or that oscillations can be detected and suppressed readily. Some instability modes may be acceptable with no potential for SAFDL violation (e.g., bi-stable flow or small flow oscillations during low-pressure startup).

4.A.1.2 Summary of Technical Information

To meet GDC 12, applicant used a stability criterion of DR less than 0.8 for all three density wave stability modes—corewide, channel, and regional. The applicant's criteria provide a DR margin of 0.2 to the ultimate criteria of a DR less than 1.0 to account for uncertainties. In addition, applicant calculated the uncertainties in its best-estimate DR calculation using the code, scaling, applicability, and uncertainty (CSAU) methodology. Applicant applied this uncertainty to the DR calculated for normal operation with a feedwater temperature of 216 degrees C (420 degrees F) (point SP0 on SER Figure 15.1-3), effectively accounting for uncertainties twice.

ESBWR-specific analyses demonstrate that unstable power oscillations are highly unlikely, thus complying with GDC 10. These calculations will be performed on cycle-specific bases as part of the reload analysis procedures to confirm the stability of the ESBWR for future cycles. As backup protection, the ESBWR design implements a defense-in-depth (DID) D&S solution based on the approved detect and suppress solution – confirmation density (DSS-CD) solution documented in NEDC-33075P-A, "General Electric Boiling Water Reactor Detect and Suppress Solution—Confirmation Density."

4.A.1.2.1 Density Wave Stability Results

In ESBWR DCD Tier 2, Section 4.D, NEDO-33337, and NEDO-33338, applicant presented the results of its stability analyses. To prevent density wave instabilities, applicant designed the ESBWR to have a very low DR during AOOs Tables 4D-2 through 4D-4 of the ESBWR DCD present the DRs for channel, super-bundle (16 fuel bundles), core, and regional oscillations for rated feedwater temperature operating conditions in an equilibrium core. NEDO-33337 and NEDO-33338 document the stability performance for the initial core at both rated and off-rated feedwater temperature conditions. This analysis demonstrated that the most limiting stability condition corresponds to the reduced feedwater temperature point (SP1M). The minimum allowable feedwater temperature for point SP1M (on SER Figure 15.1-3) is confirmed on cycle-specific bases to ensure that the calculated DR is less than 0.8 following a loss of feedwater heating anticipated occurrence from point SP1M.

4.A.1.2.2 Nondensity Wave Instabilities

Applicant identified two potential nondensity wave mechanisms for flow oscillations at low pressure (i.e., during startup). 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 the 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 nondensity 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 unavoidable in a natural circulation reactor because this instability region must be crossed before a steady two-phase voided region is established in the chimney. Applicant stated that the magnitude of these oscillations is small and the margin to critical power is very large; thus, these oscillations have no potential to violate SAFDLs and are acceptable under GDC 12.

In response to staff questions, applicant also evaluated the loop-type instability during normal operations by perturbing the chimney void fraction. Applicant showed that flow oscillations develop between the chimney and the downcomer, but they are highly damped, indicating that the ESBWR is not susceptible to oscillations from this mode that could potentially exceed SAFDLs. For these calculations, a fine chimney nodalization scheme was used to minimize numerical damping.

4.A.1.2.3 Startup

In ESBWR DCD, Section 4D.2, applicant summarized a typical startup procedure and a TRACG analysis of the startup trajectory. Applicant presented the startup trajectory using an imposed core power (i.e., no neutronic feedback) with three different heatup rates. The lowest power level of 50 MW corresponds to a heatup of 30 degrees C/hour (54 degrees F/hour). Applicant stated that this is likely to be close to the actual value for startup. The median power level was 85 MW, with a corresponding heatup rate of 55 degrees C/hour (99 degrees F/hour). This is the highest allowable heatup rate to comply with reactor vessel thermal stress requirements. The highest power level applicant used was 125 MW, which corresponds to a heatup rate of 82 degrees C/hour (147.6 degrees F/hour), which is above the allowable limit. Applicant showed large thermal margins to SAFDLs for the three heatup rates. Small-amplitude oscillations develop when voiding starts at the top of the chimney; however, the core is still subcooled at that time and exhibits a large margin to CPR. Therefore, these oscillations do not have the potential of violating SAFDLs and satisfy the requirement of GDC 12.

4.A.1.2.4 Technical Specifications

TSs related to stability are part of the oscillation power range monitor, which implements the DID solution as described in Section 4D.3 of the DCD. The setpoints are cycle independent and are documented in Table 4D.5 of the DCD.

4.A.1.2.5 Analysis Methodologies

APPLICANT used the TRACG coupled thermal-hydraulics three-dimensional neutronics code to analyze stability margins. TRACG is a time-dependent code with a full two-fluid representation and explicit numerics capability. NEDC-33083P, Supplement 1, documents the TRACG04 code and APPLICANT analysis methodology for calculating stability margins in the ESBWR, and the corresponding SER presents the staff's approval. The stability analysis statistically accounts for the uncertainties and biases in the models and plant parameters using a Monte Carlo method for the normal distribution one-sided upper tolerance limit if the output distribution is normal, or the order statistics method if it is not. The application of the CSAU uncertainty methodology as it applies to stability is described in more detail in the response to RAIs 4.3-22 and 15.2-23. Based on the applicant's responses, RAIs 4.3-22 and 15.2-23 were resolved.

4.A.1.3 Staff Evaluation

The following sections document the staff's evaluation of the information presented by Applicant in ESBWR DCD Tier 2, Section 4.D, for the equilibrium core at rated feedwater temperature, and NEDO-33337 and NEDO-33338 for the initial core at both rated and off-rated feedwater temperatures. The staff followed the review procedures in SRP Section 15.9.

4.A.1.3.1 Applicability of the ESBWR Stability Criteria

Traditional BWRs (BWR/2-6) use a stability acceptance criterion on a two-dimensional map where core and channel DRs are set at limits of 0.8 and there is a cutout of the upper right corner of the defined rectangle where regional oscillations are expected to occur. This is sometimes referred to as the "dog-bite" correlation or the FABLE (a frequency domain stability code) criterion. When this criterion was established, no code was able to calculate the regional DR directly. Since TRACG is capable of predicting the regional DR, the staff requested that applicant calculate the ratio directly. Applicant implemented this change in response to RAI 4.4-10. In addition, applicant performed a Monte Carlo analysis of channel, corewide, and regional stability at rated power and flow and the limiting exposure for each stability mode. The limiting exposure is determined through iterative calculations found in Section 8.3.1 of NEDE-33083P-A, Supplement 1, "TRACG Application for ESBWR Stability Analysis," issued December 2004. Based on these calculations, the DCD reports the one-sided upper tolerance limit with 95 percent content and 95-percent confidence level, which is roughly equivalent to a 2σ statistical treatment for normal distributions. From these calculations, the staff observes that the estimated TRACG uncertainty (at the 95/95 or 2σ level) is less than 0.2. The 0.8 DR acceptance criterion allows for 0.2 in uncertainties, and applicant has demonstrated that this allowance is adequate. The acceptance criterion is conservative as both the predicted DR and the acceptance criterion itself include the uncertainties. Based on the applicant's response, RAI 4.4-10 was resolved.

4.A.1.3.2 Density Wave Stability Results

DCD Section 4D.1.3 presents the stability results calculated by TRACG for the candidate ESBWR plant design, with 1,132 bundles and a rated thermal power of 4,500 megawatts thermal, operating at rated feedwater temperature. The TRACG ESBWR model includes 24 thermal-hydraulic regions plus 4 hot channels. The TRACG core-wide model uses a different channel grouping but the regional mode results tend to be the limiting case for ESBWR stability evaluations.

Analysis was conducted at various points of an equilibrium GE14E cycle: BOC, MOC at the peak hot excess (PHE) reactivity point, and EOC. The predicted DRs under steady-state conditions for ESBWR using TRACG are well within the acceptance criteria (DR less than 0.8). The DRs calculated by staff confirmatory LAPUR calculations are similar, and range from 0.12 to 0.24. These DRs are very small (very stable conditions) and hard to estimate accurately. One-to-one comparisons between calculations are not possible because the DR “estimation” error dwarfs all other errors at these low values. The conclusions of this review are based on the fact that both TRACG and LAPUR predict similarly low DRs at the rated feed water temperature.

Applicant also conducted analyses for an initial core at off-rated feedwater conditions. The analyses indicate that low feedwater temperatures result in lower margin to stability. This is caused by a shift to the bottom of the axial power shape.

Indeed, as described in NEDO-33338, stability considerations limit the minimum feedwater temperature allowed for operation. The feedwater temperature of point SP1M (on SER Figure 15.1-3) is defined so that the DRs calculated following a loss of feedwater heater transient are less than the 0.8 criteria. This calculation is performed on a cycle-specific basis and the minimum allowed feedwater temperature is reported in the COLR.

Stability is a crucial design requirement for the ESBWR because the rated power and flow conditions are the limiting conditions for stability during normal operation. However, following an AOO, the power/flow conditions could be even more severe than at rated conditions; therefore, AOO analyses must include an evaluation of stability. In general, the stability margin reduces when the reactor power increases and/or core flow reduces. Because the ESBWR design relies on natural circulation for core flow circulation, the core flow during full-power operation depends only on the vessel water level. Higher water level means higher core flow and vice versa. During normal operation, the water level is tightly controlled, and a reactor scram is initiated when the water level is too high or too low.

The DCD identifies two AOOs with the potential to decrease the ESBWR stability margin: LOFWH, which results in increased power; and loss of feedwater flow (LOFW), which results in a lower flow. DCD Table 4D-4 shows the DRs calculated by TRACG for this event when the ESBWR is operating at rated feedwater temperature; the most limiting event is the increase in power caused by the LOFWH. The core DR increases by about 0.14, but it remains well below the acceptance criteria. The LOFW is a milder event because the scram system trips the reactor when the water level reaches the level 3 setpoint. Operation with reduced feedwater temperature results in a decrease of stability margin, and the LOFWH from point SP1M is the limiting stability event, with a calculated regional DR of 0.71 (see Table A.1-3 of NEDO-33338). This calculation assumes a reduction of 16.7 degrees C (30 degrees F) in feedwater temperature; larger temperature reductions would result in SCRRI initiation and suppression of the event.

The staff review concurred with applicant's evaluation of the effects of AOOs on ESBWR stability margins. The results meet the acceptance criteria discussed in Section 4.A.1.1 of this report, and the calculations show that the ESBWR is stable under the postulated AOOs.

In RAI 4.4-57, the staff requested that applicant provide regional mode DRs for the two limiting AOOs. By letter dated March 10, 2008, the applicant submitted MFN 08-224, which contains the response to RAI 4.4-57 and an evaluation of regional DRs during AOOs. The applicant also updated Table 4D-4 of the DCD on the basis of these results. The analyses indicate that the regional DR is limiting for AOOs. Based on the applicant's response, RAI 4.4-57 was resolved.

4.A.1.3.3 Nondensity Wave Instabilities

The staff reviewed the potential for nondensity wave instabilities in the ESBWR. Considering the startup instabilities identified by applicant (geysering and Type 1), the staff agreed that Type 1 instabilities will occur during startup. However, these will not pose a challenge to SAFDLs because of the large margins and low power during startup, and therefore the staff found that these instabilities are acceptable during startup and are not inconsistent with GDC 12. Section 4A.1.3.4 of this report discusses the startup.

The staff also considered the potential for loop-type (or buoyancy-driven) oscillations during normal operations. The staff requested that applicant perturb the buoyancy term in the chimney to confirm that these oscillations do not develop. A fine nodalization scheme was used for these calculations to avoid numerical damping. The oscillations damped immediately. In RAI 4.4-58, Supplement 1, the staff requested the applicant to explain the apparent differences between TRACG04 results and experimental results in the GENESIS facility. Applicant's response to this RAI indicates that TRACG04 reproduces the GENESIS experimental results when the neutronic feedback in TRACG04 is turned off, simulating the electrically heated bundles in the experimental facility. In its response to RAI 4.4-58, Supplement 1, and RAI 4.4-11, the applicant provided data supporting the assertion that the ESBWR chimney has no significant effect on stability. These chimney results are independent of the chimney nodalization (coarse or fine). The results of these calculations show that loop oscillations driven by chimney buoyancy perturbations are not likely to develop in the ESBWR. Based on the applicant's responses, RAIs 4.4-58 and 4.4-11 were resolved.

If the ESBWR is operated close to a flow-regime transition boundary, it is conceivable that an oscillatory instability may develop. The staff considered the potential for flow-regime transition instabilities to develop in the ESBWR. At rated power, the ESBWR is expected to have fully developed churn-turbulent flow, except for possibly a few low-power periphery partitions. In addition, there will be thermal-hydraulic communication between all of the chimney partitions and channels via the core bypass, which will tend to equalize the partition void conditions. In response to RAI 4.4-39, applicant confirmed that the pressure at the outlet of the core will be uniform across the core. This is because the core outlet (and chimney inlet) conditions communicate hydraulically via the liquid level in the core bypass. The staff issued a supplemental RAI requesting applicant to evaluate the bypass flow conditions. The staff concluded that flow regime transition oscillations will not be a concern in the ESBWR at rated conditions. The staff disagreed with the applicant's assertion that the TRACG and PANACEA calculations are independent based on information provided in the response to RAI 21.6-85. In RAI 4.4-39 S02, the staff requested that the applicant perform an analysis to determine the core outlet pressure distribution using an independent verification approach. Applicant provided a TRACG calculation that uses an initialization process that is independent of PANACEA. The results of this calculation confirm that core outlet pressure is uniform. Based on the applicant's response, RAI 4.4-39 was resolved.

In RAI 14.2-89, the staff requested that, during startup testing, the COL holder characterize the power levels at which flow-regime transition oscillations may possibly occur. The staff recommended that the licensee analyze the neutron flux from LPRMs under each chimney partition. Also, the staff requested that applicant develop a startup testing plan to identify the impact, if any, of operation at reduced power levels where oscillations induced by flow transition may be possible.

In the response to RAI 14.2-89, the applicant revised the DCD Revision 5, Section 14.2.8.2.7 identifying this test as a initial test program (ITP) in order to identify the impact of any possible flow oscillations and committed to developing an additional single plant startup test based on LPRM readings. This ITP is also identified in the DCD Tier 1, Section 3.5, and represents a commitment that combined operating license applicants referencing the certified design will implement an ITP that meets the objectives presented above. Based on the applicant's response, RAI 14.2-89 was resolved.

In response to RAI 21.6-113, the applicant argued that chimney entrance effects and flow at the chimney inlet that is not fully developed could have two separate effects: (1) alteration of the steady-state void fraction or (2) induction of time-dependent fluctuations (i.e., noise) in the void fraction. The applicant evaluated the steady-state void effects and concluded that the real void fraction at the chimney inlet may be lower than that calculated by TRACG04. The applicant presented experimental evidence suggesting that the length of the entrance region is small (approximately one equivalent diameter) relative to the chimney height such that the effect on the calculated chimney static head and natural circulation flow is small. The experimental evidence includes (1) data from the Dodeward reactor and (2) data from Dubrovskii, which covers the reactor operating pressure (75 bar) and has a similar diameter (0.61 meters). Based on these data, the applicant concluded that the steady-state void fraction at the chimney inlet could be as low as 75 percent of the fully developed void, but the region not fully developed is at most 1 meter long. The applicant performed a calculation assuming 70 percent of the fully developed void for 1 meter and determined that the impact on recirculation (core) flow is less than 3 percent. This 3-percent overprediction is an upper-bound estimate, and the actual flow error is expected to be smaller.

The staff concurred with the applicant's evaluation. The available data indicate that entrance effects and not fully developed flow may reduce the steady-state void fraction by up to 25 percent, but only for the first meter of chimney. The effect on recirculation core flow of this misprediction is small (less than 3 percent) and should not have any significant effect on TRACG calculations.

The applicant presented experimental evidence of the impact of time-dependent void fraction fluctuations (i.e., noise), especially in the churn-turbulent regime. The data presented include an evaluation of the Dodeward reactor data and the Dubrovskii data. The applicant reported that no significant flow oscillations were observed in the experimental data. Thus, the applicant concluded that void fraction oscillations caused by turbulence in the churn flow regime will have little or no effect because if they are fast, they will be averaged out in the chimney. If they are slow, they will be compensated for by changes in core exit void to maintain the reactor critical. In addition, both the Dodeward and Dubrovskii experimental data indicate negligible flow oscillations. Both sets of experimental data are in the churn-turbulent flow regime. The staff concurred with the applicant's evaluation. Based on the applicant's response, RAI 21.6-113 was resolved.

The staff reviewed the information presented by the applicant related to flow oscillations in the ESBWR chimney. This information included (1) TRACG04 calculations with a detailed axial nodalization, (2) a TRACG04 benchmark against the GENESIS experiment, and (3) an evaluation of the chimney entrance effects and flow regime oscillations using experimental data. The staff concurred with the applicant's evaluation that loop oscillations driven by chimney buoyancy perturbations are not likely to develop in the ESBWR.

4.A.1.3.4 Startup

During normal operation, the stability mode of concern is the so-called density wave that produces flow and power oscillations within a frequency range between 0.5 and 1 hertz (Hz). Because of its unique startup process, other instability modes are of concern during ESBWR startup. These instability modes include geysering instability and loop instabilities (also known as manometer or Type I instabilities). The TRACG capability of modeling both of these modes was reviewed and accepted in NEDE-33083PA, Supplement 1.

The key in the startup procedure is maintaining power low enough so that boiling occurs only at the top of the chimney and not inside the active core. By maintaining voids out of the core at low pressure, the ESBWR prevents reactivity feedback issues, which could result in violent power oscillations.

As the circulating water is slowly heated, saturation temperature is first reached at the top of the separators because the pressure is lower, given the density head or weight of the column of water in the chimney. Vapor generation at the top of the separators results in a reduction in the chimney density, which reduces the pressure causing the voiding front to propagate downward. The formation of voids also results in a larger driving head for natural circulation flow. The increase in natural circulation flow reduces the core exit temperature and leads to a collapse of the voids. This completes one cycle of the hydrostatic head oscillation, and these oscillations persist until the temperature of the water inventory in the core increases and a steady void fraction is established in the separators. Small oscillations in the flow rate are harmless when the power is low and the core flow is single phase, and consequently, thermal limits have a very large margin.

Applicant simulated the ESBWR startup procedure with TRACG and demonstrated that the ESBWR proposed startup procedure is feasible. The results showed no significant power oscillations even for heatup rates larger than allowed by TSs. CPR limits were not violated by any of these scenarios.

Oscillations do develop during the startup as Type 1 (manometer type) instability. These oscillations can be seen as a rapid variation of void fraction in the separators. Because the core coolant is subcooled at the time of the oscillations, the margin to boiling transition is very large. Flow oscillations in subcooled regimes are of no consequence to the SAFDLs. GDC 12 specifies that "The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed." Since the predicted Type 1 ESBWR instabilities have no potential to exceed SAFDLs, the staff concludes that their presence is not inconsistent with GDC 12, and these oscillations are acceptable.

In RAI 4.4-59, the staff requested that applicant "establish a maximum heat-up rate for the low pressure start-up in terms of MW per hour that will not be exceeded by the licensee. Show margin to instability by simulating the start-up path using a larger heat-up rate that bounds the established maximum. Use neutronic feedback."

In response to RAI 4.4-59, the applicant performed a detailed analysis of the ESBWR startup with a wide range of parameter variations to attempt to bound the expected startup conditions. For all these simulations, neutronic feedback has been simulated, as requested. The study concludes that startup rates as high as 110 degrees C/hour (198 degrees F/hour) are safe and free from instabilities that could challenge SAFDLs. This demonstrated safe value is twice as large as the maximum heatup rate allowed by the thermal-stress limit of 55 degrees C/hour (99 degrees F/hour), and approximately four times larger than the expected ESBWR heatup rate of 27.5 degrees C/hour (49.5 degrees F/hour). Based on the applicant's response, RAI 4.4-59 was resolved.

In the SER for NEDE-33083P, Supplement 1, the staff noted that for the ascension to full-power phase of startup, which occurs approximately 8 hours into the startup, the current approach for modeling stability analyses does not include a balance-of-plant model.

As documented in the SER for NEDE-33083P, Supplement 1, the feedback from steam flow into the feedwater system would be necessary in order to perform best-estimate analyses of the transient response to an oscillation over long time periods. In addition xenon (Xe) will have a more pronounced effect on the power distribution. In RAI 4.4-60, the staff requested that applicant provide a calculation demonstrating margins with transient Xe and include a balance-of-plant model for the startup through ascension to full power.

In response to RAI 4.4-60, the applicant did not perform the requested TRACG calculation. Instead, a series of "PIRT46" TRACG calculations was used to simulate the Xe effect via the impact on local power peaking.

The applicant presented a series of physical arguments to justify why the constant-Xe calculation is acceptable. The basis for these arguments is that a typical startup from cold shutdown to full pressure takes about 5 hours, and Xe burnup is not significant at less than 2-percent power for 5 hours.

In the response, the applicant stated that TRACG does not calculate time-varying Xe. It uses a constant cross-section set generated by PANAC11 for a given Xe condition. TRACG does not calculate time-varying Xe, but in the "PIRT46" parameter, it provides a capability to simulate Xe effects by increasing or decreasing local power peaking. The applicant performed a PANACEA study for the ESBWR initial core at MOC. PANACEA, being a series of steady-state calculations for the startup path, can model the Xe burnup. Based on these PANACEA calculations, the applicant concludes that a radial peaking factor (RPF) of 8 conservatively bounds the expected radial peaking when Xe burnup is accounted for (the nominal RPF value is approximately 5). For the TRACG calculations, RPF values as high as 11 were used. In the nominal case, an RPF value of 5 is used, which corresponds to a hot channel power of 479 kilowatts (kW) (for a heating rate of 90 MW for the core). In the Xe burnup bounding simulation, the RPF was increased to 11 (hot channel power 1,440 kW) and the resulting minimum CPR was reduced from 7.2 to 5.3. A CPR margin of 5.3 is a very significant margin. Thus, the staff concludes that Xe burnup effects are not likely to invalidate the conclusion that SAFDLs will not be exceeded during startup. Thus, the GDC 12 requirements are satisfied even when Xe burnup is accounted for. Based on the applicant's response, RAI 4.4-60 was resolved.

4.A.1.3.5 Effect of Chimney Models

In response to staff RAIs, applicant performed a series of detailed analyses of the effect of the chimney on the density wave and loop stability modes. The ESBWR TRACG model was modified to include a fine node structure in the chimney region. The analyses included a case with a corewide power response to a pressure perturbation and cases with buoyancy perturbations. The staff concluded that the finely nodalized chimney allows for a more accurate representation of void propagation through the chimney but has no effect on the stability results. Even though applicant stated that the original nodalization used for the stability calculations in the DCD are adequate for stability analyses, the staff recommended in RAI 4.4-58 that the TRACG model with the fine chimney nodalization be used for future ESBWR stability calculations. Applicant responded to this RAI by stating, "In summary, the finely nodalized chimney allows for a more accurate representation of void propagation through the chimney, but has no effect on the stability results. The original nodalization used for the stability calculations in Reference 4.4-11-1 and the DCD is adequate for stability analysis." The calculation discussed in the response to RAI 4.4-58 applies to the ESBWR and shows that results are insensitive to the nodalization model. Therefore, applicant does not believe it is necessary to perform stability calculations in support of the DCD with the fine nodalization chimney model of TRACG to guarantee that chimney oscillations do not affect the core stability.

The staff issued a supplement to RAI 4.4-58, which explains an apparent incompatibility of results between TRACG calculation and experimental data from the GENESIS facility. GENESIS is a thermal-hydraulic loop simulation of the ESBWR with a single channel and a long chimney. The power to the channel may be modulated by a computer-simulation of the reactivity feedback based on online void fraction measurements. In the GENESIS facility experiments, a low frequency of oscillation (approximately 0.1 Hz) was observed when the power to the channel was maintained constant. This is an approximation of the purely thermal hydraulic or "channel" oscillation mode simulated by TRACG. The TRACG results did not agree with the experimental data and showed a significantly larger oscillation frequency (approximately 0.8 Hz). These results indicated that the chimney did not take part in the TRACG oscillation, while the GENESIS results indicate that the chimney does take part in the oscillations because of the lower oscillation frequency.

In the response to RAI 4.4-58, S01, the applicant performed a TRACG04 simulation where the chimney buoyancy term was perturbed. An oscillation of about 0.1 Hz was observed when the channel power was maintained constant, simulating the GENESIS results. The applicant concluded that there is no discrepancy between the GENESIS and TRACG04 results and that TRACG04 can model loop-type oscillations in the chimney.

In the second part of the response to RAI 4.4-58, S01, the applicant justified the use of coarse nodalization in the chimney. The applicant argued that the chimney does not play an important role in the density wave instabilities of interest. Loop oscillations (where the chimney plays an important role) are not limiting in the ESBWR and do not pose any significant safety concern. The applicant concluded that the coarse chimney nodalization was adequate for ESBWR stability analysis.

After review of the available data, the staff found that (1) when using fine nodalization, TRACG can model the loop-type buoyancy-driven flow oscillations that were observed in the GENESIS experiment, (2) both TRACG04 and GENESIS are in relatively good agreement in predicting the frequency and DR of chimney loop-type oscillations, and (3) for the density wave oscillations that are likely to be limiting in the ESBWR, the chimney does not appear to play a significant dynamic role, and thus, numerical damping in the chimney region is not likely to affect the magnitude of the calculated DR.

Therefore, the staff concurred with the applicant's evaluation and accepted that a coarse chimney nodalization would be sufficient to model density wave oscillations. Based on the applicant's response, RAI 4.4-58 was resolved.

4.A.1.3.6 Stability Long-Term Solution

ESBWR DCD Tier 2, Revision 3, Section 4.3.3.6.2, "Thermal Hydraulic Stability," indicates that a D&S solution is the preferred option for the ESBWR and that all operating BWRs are required to implement an approved long-term stability solution.

In RAI 4.3-7, the staff requested applicant to provide a detailed description of the stability solution chosen for the ESBWR, whether it needs further staff review or it is a standard solution, associated TSs, and how the TSs reflect the setpoint calculation (if any).

Applicant responded that it has selected the standard D&S solution DSS-CD, as documented in the NRC-approved GE proprietary report NEDC-33075P, Revision 5, dated November 2005. Applicant also provided a proposed DCD revision to incorporate DSS-CD into the ESBWR TS Sections 3.3.1.4 and 3.3.1.5 with cycle-specific setpoints for the DSS-CD system to be provided in the individual plant COLR, as specified in TS 5.6.3. In addition, the applicant updated the DCD in Revision 5 to include Section 4D-3, which describes the ESBWR specific features of the DID D&S solution proposed for the ESBWR. This solution uses all the approved algorithms from DSS-CD, with parameter settings adjusted to the special ESBWR characteristics. Because it is a DID measure, a licensing-basis calculation is not required to demonstrate the effectiveness of the solution in preventing SAFDLs. The staff concluded that the ESBWR DID solution satisfies the requirements of GDC 13, 20, 29 and GL 94-02 because the ESBWR DID is implemented in the already approved DSS-CD and Option III hardware, which satisfies the instrumentation and controls and hardware requirements of GCD 13.

2. The ESBWR DID is a defense-in-depth feature, but in the case oscillations were to develop, it would initiate an automatic scram, which satisfies the requirements of GDC 20 and 29.

3. The licensing basis of the ESBWR is demonstration of stability by analysis; therefore, the ESBWR implements a Solution I type of LTS. Through TRACG04 analyses, ESBWR operators will demonstrate on a cycle-specific basis that the ESBWR will always operate outside of the stability exclusion region. In addition, the DID solution will provide a D&S (Solution III type) feature as DID. A backup stability solution is also provided as required by GL 94-02. Thus, the staff concludes that the ESBWR stability methodology satisfies the requirements of GL 94-02.

Based on the applicant’s response, RAI 4.3-7 was resolved.

In RAI 4.3-8, the staff pointed out that all approved D&S solutions have an armed region. Typically, the solution is only armed for low-flow maneuvers and represents a small fraction of the cycle time. Since the ESBWR operates at the equivalent of low-flow at nominal conditions, one expects that the D&S solution must remain armed for the complete cycle. The staff requested applicant to discuss the armed region implications and the associated probability of false alarms.

In the response to RAI 4.3-8, the applicant described the stability LTS armed region. The armed region will include normal operation, and it will be defined in the COLR based on power and feedwater temperature. Based on the applicant’s response, RAI 4.3-8 was resolved.

In RAI 4.3-9, the staff indicated that a future licensee may have the flexibility to deviate from the standard certification and choose a different long-term stability solution. The staff requested applicant to specify criteria that must be met by the ESBWR for long-term stability solutions.

In the response to RAI 4.3-9, and referenced in DCD Tier 2, Revision 7, Chapter 4, Section 4D and Chapter 16, Specification 3.3.1.4, the applicant stated the requirements that a future LTS must meet, commitments that (1) GDC 12 will be satisfied (2) the LTS will provide a backup solution in case the primary is declared inoperable and (3) the backup solution will not be active for longer than 120 days. Based on the applicant’s response, RAI 4.3-9 was resolved.

4.A.1.3.7 Analysis Methodologies

Applicant used the TRACG04 code and methodology as documented in NEDE-33083P, Supplement 1, to calculate stability margins for the ESBWR. The staff reviewed this methodology during the pre application phase of the ESBWR and accepted it with open items. The staff review of the open items for this method is documented as an addendum to the SER for NEDE-33083P and is also summarized in Section 21.6 of this report.

For analysis in support of the ESBWR DCD, the applicant used three different versions of the TRACG04 code to generate stability results. The primary reason for the use of the different code versions was the date of the calculation. All code versions were "non-level 2" and were used only after a validation of the code against experimental data previously performed. These "external-data validations" served the purpose of alternate calculations as required by the applicant's Engineering Operation Procedure (EOP) 42-6. The applicant used the following TRACG04 versions:

4. T4N2, which corresponds to version 45 in the Alpha platform, this version was used for the early DCD calculations. Three exposures were calculated (BOC, MOC, and EOC). In addition, the original "stability during startup" calculation was performed with this version assuming a constant power generation (no neutronic feedback).

5. T4N3, which corresponds to version 49 in the Alpha platform, this version was used to respond to staff RAIs related to the earlier T4N2 calculations. The stability during startup calculation was also updated including three-dimensional neutronic feedback, as requested by the staff. Version 49 was used for this calculation because it was the most recent, validated version of TRACG at the time.

6. T4PN53, which corresponds to version 53 in the PC platform, this version was used to respond to staff RAIs related to the earlier T4N2 calculations. In response to these RAIs, the applicant performed a number of stability calculations around the MOC point with a fine mesh of exposures to identify the maximum DR as function of exposure. The PC V53 version was used because (1) it was the most recent version and (2) the PC version is significantly faster and allowed to perform the many calculations required to step through the exposure fine mesh to identify the maximum DR.

4.A.1.3.8 Staff's Independent Calculations

The staff performed independent calculations using the LAPUR code to evaluate the stability of the ESBWR. LAPUR is a frequency domain code developed by Oak Ridge National Laboratory that is used for BWR stability analysis. The staff performed calculations at 12 points of a representative fuel cycle at nominal feedwater temperature using the design information in the ESBWR DCD. The LAPUR confirmatory calculations showed that the ESBWR stability is within the limits of the design criteria.

The highest calculated DR at nominal feedwater temperature is 0.24, and corresponds to the corewide stability mode for the EOC conditions, when the axial power shape becomes flat or almost top peaked. The LAPUR results were in good agreement with the ODYSY and TRACG results reported by the applicant for the nominal operating conditions. The LAPUR confirmatory calculations also indicated that the dynamic model used to simulate the chimney riser has little or no effect on the stability of the ESBWR. The riser itself has a large effect on the core flow, but it has a very small friction pressure drop. However, once the core flow and power are fixed, the presence of the chimney does not influence stability. Accordingly, the chimney plays a crucial role in setting up the steady-state value of the core flow, but plays only a minor role during the unstable oscillations. As a result of its calculations, the staff concludes that the ESBWR DR is within the limits of the acceptance criteria.

4.A.2 ESBWR Stability during Anticipated Transient without Scram

Chapter 15 of this report contains the major part of the review of the ATWS event scenario. This section addresses the issue of thermal-hydraulic stability during an ATWS scenario.

4.A.2.1 Regulatory Criteria

The staff based its review of ESBWR stability performance during an ATWS event on SRP Section 15.8, "Anticipated Transients without Scram," which states the following procedure for BWR ATWS/stability reviews:

For BWRs, the ATWS/stability evaluation was addressed generically in topical reports NEDO-32047 and NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," issued October 2009, which defines the ATWS mitigation actions for plants operating up to original licensed thermal power. SRP Section 15.8 gives the following guidance:

7. For all applications, the reviewer will evaluate the implementation of the ATWS/Stability Mitigation Actions in design-specific EPGs [Emergency Procedure Guidelines], or plant-specific EOPs or EOIs [Emergency Operating Instructions]. The reviewer will ensure that sufficient information has been provided to justify that the mitigation actions are effective in maintaining core coolability criteria for the limiting ATWS/Stability event.

8. For BWRs that implement, extended power uprate (EPU) and expanded power-flow domains (e.g., MELLLA+), the licensee will demonstrate that the ATWS/Stability Mitigation Actions are effective in maintaining core coolability criteria for the limiting ATWS/Stability event.

9. For evolutionary BWRs, the licensee will provide EOPs or EOIs that implement ATWS/Stability Mitigation Actions equivalent to those approved in Reference 8 [NEDO-32047 and NEDO-32164], including manual boron injection if oscillations are detected. The licensee will demonstrate the EOPs or EOIs are effective in maintaining core coolability criteria for the limiting ATWS/Stability event.

4.A.2.2 Summary of Technical Information

To demonstrate that there are no stability issues during an ATWS transient for the ESBWR, applicant stated that the ATWS mitigation features for the ESBWR include automatic feedwater runback and automatic boron injection.

Applicant simulated an ATWS event for the MSIV closure event using TRACG04. In its response to RAI 21.6-45, applicant described the method used to perform this calculation. During this event, applicant introduced a flow perturbation at the inlet of the channels during the transient and showed that the ATWS acceptance criteria are satisfied even though a small-amplitude power oscillation was observed. Based on the applicant's response, RAI 21.6-45 was resolved.

4.A.2.3 Staff Evaluation

To demonstrate acceptable performance, a limiting ATWS scenario must be investigated. In this case, the limiting ATWS scenario is that which leads to the greatest magnitude oscillation and it is treated as a separate requirement from those dictated by SRP Chapter 15. The staff agreed with the applicant in the selection of the MSIV closure ATWS event as the limiting event for the Chapter 15 analysis, as this particular event simultaneously challenges the system integrity with high neutron flux, high vessel pressure, and high suppression pool temperature.

However, the staff had previously reviewed TRACG calculations of ATWS instability events for operating BWRs and determined that conditions exist for particular ATWS scenarios where instability events are, in fact, likely. These scenarios are those that result in high power and low flow. An isolation event such as MSIV closure will result in a rapid increase in reactor pressure, which leads to the actuation of the DPVs. The ensuing depressurization reduces reactor power. For a conventional BWR, because of the SRVs, instabilities are more likely to occur when the system is not isolated.

Again for conventional BWRs, when recirculation pumps trip (reducing flow) or on a loss of feedwater heat (increasing power and shifting power towards core bottom), the system becomes more susceptible to thermal-hydraulic instability as there is a high power-to-flow condition following in either of these events. The downward shift in axial power following an LOFWH reduces the single-phase to two-phase pressure drop ratio, thereby further reducing the stability margin. Similarly, turbine trip with full bypass may produce a pressure perturbation that will impact core reactivity by collapsing voids at the initiation of the transient, yet not initiate an isolation of the RPV. Therefore, the analysis of ATWS stability should be addressed using a limiting transient from the perspective of core stability.

The ESBWR ATWS mitigation actions include the following:

10. A reliable RPS with two redundant methods of inserting control rods: (a) hydraulic rod insertion and (b) electrical FMCRD insertion. By reducing common-cause failure mechanisms, these redundant systems make the probability of failure to scram small.

11. An alternate rod insertion function, which uses sensors and logic that are diverse and independent of the RPS, as required by the ATWS rule.

12. Automatic feedwater runback, which reduces the reactor water level and the core power generation. This function is a substitute for the recirculation pump trip required by the ATWS rule.

13. Automatic initiation of standby liquid control, as required by the ATWS rule for new reactors.

The ESBWR hardware design described in actions (1) and (2) above reduces the probability of a failure to scram. Actions (3) and (4) are an implementation of the ATWS/stability mitigation actions in operating reactors. The ESBWR implements automatically the EPG ATWS/stability mitigation actions without the need for operator intervention. With these mitigating actions, the ATWS/stability event will not be allowed to progress, and large-amplitude unstable power oscillations are not likely to develop in the ESBWR.

As the ESBWR design does not include recirculation pumps, the staff requested that the applicant select the LOFW accident and turbine trip with full bypass as the events for predicting the system performance during an ATWS instability event. Each event is a nonisolation event resulting in increased reactor power. In the case of LOFWH, the increased reactor power comes from an increase in coolant subcooling and hence an increase in moderator density. In the case of a turbine trip with full bypass, a momentary pressure wave sent down the steamline leads to a momentary reduction in core void content. In RAI 21.6-51, the staff requested that applicant use the approved methodology in the SER for NEDE-33083P, Supplement 1, to perform a DR calculation or to add margin by increasing the void reactivity coefficient.

In response to RAI 21.6-51, the applicant concluded that the most limiting ATWS event from the point of view of stability is turbine trip with bypass (TTWB). The applicant argued that TTWB is more limiting than loss of feedwater flow (LFWF) because LFWF only reduces the water level, while TTWB reduces the water level (because of the feedwater runback) and increases the subcooling significantly. Nevertheless, in spite of this evaluation, the applicant presented results for both the TTWB and LFWF.

The ATWS/stability evaluation was performed using a regional-mode channel grouping scheme, which does not preclude corewide oscillations and is, therefore, more general. In addition, a 130-percent multiplier was added to the density reactivity coefficient to increase the conservatism. Figure A.4.2.1-1 of NEDO-33338, Revision 1, shows the result of this calculation. The TRACG04 ATWS analysis shows that, in all of these conservative calculations, the ESBWR is slightly unstable under ATWS conditions because a small-amplitude regional limit cycle is observed early in the transient. Approximately 85 seconds into the transient, the automatic water-level reduction results in uncovering of the steam separators, and the self-sustained limit cycle oscillation decays as the ESBWR becomes once again stable without operator intervention. The TRACG04 calculations indicate that CPR or other limits were not violated during this bounding ATWS transient.

The staff concludes that the ATWS criteria are satisfied even though a small-amplitude power oscillation was observed. Based on the applicant’s response, RAI 21.6-51 was resolved.

In conclusion, the staff finds that large-amplitude unstable power oscillations (ATWS/stability) that could compromise ATWS criteria are not a likely event in the ESBWR because (1) the ATWS/stability mitigation actions are implemented automatically and (2) the low probability of a failure to scram.

4.A.3 Summary of Evaluation Findings

A summary and the major conclusions from the staff’s review are provided below:

(1) The stability criteria set forth in the ESBWR DCD comply with the guidelines in SRP Section 15.9. The acceptance criteria for calculated DRs for the three density wave instability modes are the following:

   a. channel DR less than 0.8
   b. corewide DR less than 0.8
   c. regional DR less than 0.8

(2) The ESBWR DRs will be calculated using TRACG and the methodology documented in NEDE-33083PA, Supplement 1, or an alternate methodology which has been reviewed and approved by the staff for use in ESBWR applications.

(3) The ESBWR DR values used in the acceptance criteria for the rated feedwater temperature conditions include a one-sided upper tolerance limit with 95-percent content and 95-percent confidence level. The uncertainty values are determined by a Monte Carlo analysis using the CSAU methodology. This is an acceptable treatment of uncertainties, and it is conservative because the acceptance criteria already contain a 0.2 margin to account for variability in modeling assumptions.

(4) Applicant calculations and staff confirmatory calculations indicate that the ESBWR satisfies the stability criteria at rated feedwater temperature conditions. The largest estimated DR is 0.53 (regional mode, MOC).

(5) Applicant calculations and staff confirmatory calculations indicate that the ESBWR satisfies the stability criteria at off-rated feedwater temperature conditions. The largest estimated DR is 0.61 (at point SP1M, regional mode, MOC).

(6) The two limiting AOOs are (1) LOFWH, which increases power to the scram setpoint, and (2) LOFW, which reduces core flow until the low-water-level setpoint is reached. The highest calculated DR during AOOs that start at rated conditions is 0.66, and it corresponds to the LOFWH at MOC for the corewide instability mode. An LOFWH event initiated at off-rated conditions (point SP1M) results in a DR of 0.71. These values are within the acceptance criteria.

(7) The DCD presents an evaluation of the stability during an ATWS. The staff concurs with applicant's evaluation that stability during an ATWS is not a concern in the ESBWR.

   a. The immediate water-level reduction caused by the automatic feedwater runback reduces the power and flow rate, and it exposes the feedwater to vessel steam; therefore, the large subcooling transient that causes the ATWS/stability event in operating reactors does not occur in the ESBWR.

   b. The automatic boron injection and, most important, the direct injection into the core bypass area reduce the duration of the ESBWR ATWS so that unstable power oscillations will be highly unlikely.

(8) Two types of startup instabilities have been evaluated by applicant for the

ESBWR: geysering and Type 1 (or manometer). The staff concurs with applicant's evaluation that these instabilities will occur during startup, but will not pose a challenge to SAFDLs. These types of instabilities are acceptable and are not inconsistent with GDC 12.

(9) In addition to the density wave and startup instability modes, applicant has evaluated the loop-type instability mode by perturbing the chimney void fraction at power. Flow oscillations develop between the chimney and the downcomer, but they are highly damped, showing that this oscillation mode is very stable.

(10) To evaluate the effect of the chimney, applicant has set up a TRACG model with fine nodalization in the chimney region. In this model, the Courant = 1 limit occurs in the chimney nodes. Stability evaluations with this model and previous models with coarser nodes show no significant difference. The staff confirmatory calculation using the LAPUR code confirms these results. These calculations indicate that the chimney dynamics play a very minor role in density wave oscillations.

(11) The staff performed confirmatory calculations to determine the power level at which the chimney will transition from slug/churn to annular flow. Oscillations may occur at the flow regime transition power. The staff calculations indicated that the flow regime transition will occur between 30-percent and 70-percent power. Rated conditions will have fully developed annular flow (except for, possibly, a periphery channel). Thus, the staff concludes that flow regime transition oscillations will not be a concern in the ESBWR at rated conditions.

All stability-related open items have been resolved. Based on the preceding review, the staff concludes that the plant design adequately addresses stability issues and satisfies all the criteria specified in SRP Section 15.9 and, specifically, GDC 10, 12, 13, 20, and 29; Appendix A to 10 CFR Part 50; and GL 94-02.

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