5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Introduction

The reactor coolant system (RCS) includes those systems and components that contain or transport fluids coming from or going to the reactor core. These systems form a major portion of the reactor coolant pressure boundary (RCPB). This chapter of the safety evaluation report (SER) describes the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the RCS and the pressure-containing appendages out to and including the outboard isolation valves. This grouping of components, defined as the RCPB, includes all pressure-retaining components, such as pressure vessels, piping, pumps, and valves, which are part of the RCS or connected to the RCS. The RCPB includes any components up to and including the following:

- the outermost containment isolation valve in piping that penetrates containment
- the second of the two valves normally closed during normal reactor operation in system piping that does not penetrate containment
- the RCS safety/relief valve (SRV) and safety valve (SV) piping and the depressurization valve (DPV) piping

Section 5.4 of this report addresses various closely allied subsystems of the RCPB. The DPVs are part of the automatic depressurization system (ADS) function of the emergency core cooling system (ECCS) discussed in Section 6.3 of this report. The nuclear boiler system (NBS) pressure relief system protects the RCPB from damage resulting from overpressure. To protect against overpressure, pressure-operated SRVs and SVs discharge steam from the NBS to the suppression pool or to the drywell. The pressure relief system also acts to automatically depressurize the NBS in the event of a loss-of-coolant accident (LOCA) in which the high-pressure makeup of the feedwater (FW), isolation condenser (IC), and control rod drive (CRD) systems fail to maintain the reactor vessel (RV) water level. Depressurization of the NBS by actuation of the SRVs, SVs, and DPVs allows the gravity-driven cooling system (GDCS) to supply cooling water to adequately cool the fuel in the core. Section 5.2.5 of this report specifies the limits on NBS leakage inside the drywell so that operators can take appropriate action to prevent impairment of the integrity of the NBS process barrier.

Section 5.3 of this report describes the RV and appurtenances. The major safety consideration for the RV is its ability to function as a radioactive material barrier. The vessel design considers various combinations of loading. The design process considers the possibility of brittle fracture; addresses suitable design, material selection, and material surveillance activity; and establishes operational limits that avoid conditions in which brittle fracture is possible.

The RCS provides coolant flow through the core by natural circulation within the RV. The core coolant flow rate changes with reactor power output. The control rods are adjusted either manually or automatically with the fine motion CRDs to adjust reactor power.

The natural circulation within the RV eliminates the need for a recirculation system. Therefore, there are no large piping connections to the RV below the core, and there are no recirculation pumps.

Venturi-type main steamline (MSL) flow restrictors are part of the main steam nozzle on the reactor pressure vessel (RPV). The restrictors are designed to limit the loss of coolant resulting from an MSL break inside or outside the containment. The restrictors limit the reactor depressurization rate to a value that will ensure that the steam dryer and other reactor internal structures remain in place and limit the radiological release outside of containment before closure of the main steam isolation valves (MSIVs).

Two isolation valves are installed on each MSL. One is located inside the containment and the other is located outside the containment. If an MSL break were to occur inside the containment, closure of the isolation valve outside the containment isolates the containment. The MSIVs automatically isolate the RCPB when a pipe break occurs outside containment. This action limits the loss of coolant and the release of radioactive materials from the NBS.

The CRD system high-pressure makeup provides water by means of the reactor water cleanup/shutdown cooling (RWCU/SDC) piping to the core any time FW flow is unavailable. The high-pressure makeup mode starts automatically upon receipt of a low reactor water level signal; however, the operator can also start it manually. Section 4.6 of this report discusses the CRD system.

The RWCU/SDC system and the isolation condenser system (ICS) can be used to cool the NBS under a variety of situations. During normal shutdown and reactor servicing, the RWCU/SDC system removes residual and decay heat. The RWCU/SDC system, in conjunction with the ICS, allows decay heat to be removed whenever the main heat sink (main condenser) is not available (e.g., hot standby). The ICS provides cooling of the reactor if the RCPB becomes isolated following a scram during power operations.

The ICS automatically removes residual and decay heat to limit reactor pressure when reactor isolation occurs. Over a longer duration, the ICS provides a way to remove excess heat from the reactor with minimal loss of coolant inventory, if the normal heat removal path is unavailable.

The GDCS is an ECCS for use during a postulated LOCA. The GDCS is operational at low RV pressure following pressure reduction by the ADS function of the ECCS. Section 6.3 of this report describes the operation of the GDCS and ADS. The RWCU/SDC system recirculates a portion of reactor coolant through a demineralizer to remove dissolved impurities and their associated corrosion and fission products from the reactor coolant. This system also removes excess coolant from the reactor system under controlled conditions.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance with Code and Code Cases

General Design Criterion (GDC) 1, “Quality Standards and Records,” in Appendix A, “General Design Criteria for Nuclear Power Plants,” to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” requires that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement applies to both pressure-retaining and nonpressure-retaining SSCs that are part of the RCPB, as well as to other systems important to safety. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability.

5.2.1.1 <u>Compliance with 10 CFR 50.55a</u>

5.2.1.1.1 Regulatory Criteria

The staff reviewed Section 5.2.1.1 of the design control document (DCD), Tier 2, in accordance with Section 5.2.1.1, Revision 3, of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP).

In accordance with 10 CFR 50.55a, "Codes and Standards," components important to safety are subject to the following requirements:

- RCPB components must meet the requirements for Class 1 (Quality Group (QG) A) components, as specified in Division 1, Section III, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, except for those components that meet the exclusion requirements of 10 CFR 50.55a(c)(2).

- Components classified as QG B and C must meet the requirements for Class 2 and 3 components, respectively, as specified in ASME Code, Section III.

5.2.1.1.2 Summary of Technical Information

DCD, Tier 2, Table 3.2-1, Revision 7, classifies the pressure-retaining components of the RCPB as ASME Code, Section III, Class 1 components. These Class 1 components are designated QG A in conformance with Regulatory Guide (RG) 1.26, Revision 3, "Quality Groups Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," issued February 1976. The staff evaluated the QG classifications discussed in Section 3.2.2 of this report and finds that the economic simplified boiling-water reactor (ESBWR) mechanical and pressure-retaining components in the RCPB have been acceptably classified as QG A, in accordance with 10 CFR 50.55a.

In addition to the QG A components of the RCPB, certain lines that will perform a safety function and that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified as QG B, in accordance with Position C.1 of RG 1.26, and will be constructed as ASME Code, Section III, Class 2 components. Section 3.2.2 of this report discusses the staff's review of these components and other pressure-retaining components that will be constructed to ASME Code, Section III, Class 2 and 3 specifications.

SRP Section 5.2.1.1 recommends that safety analysis reports for both construction permits and operating licenses contain a table identifying the ASME component code, code edition, and applicable code addenda for all ASME Code, Section III, Class 1 and 2 pressure vessel components, piping, pumps, and valves in the RCPB. DCD, Tier 2, Section 5.2.1.1, Revision 7, provides ASME Code edition and applicable addenda for the ESBWR design in compliance with the requirements of 10 CFR 50.55a. DCD, Tier 2, Table 1.9-22, identifies the specific ASME Code edition and addenda.

The COL applicant must ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda in effect at the time of the COL application, as endorsed in 10 CFR 50.55a. If the ASME Code edition and addenda differ from that specified in the DCD, the COL applicant should identify in its application the portions of the later ASME Code editions and addenda for NRC staff review and approval.

5.2.1.1.3 Staff Evaluation

ESBWR DCD, Revision 5, Section 5.2.1.1, does not address the ASME Code of record edition and addenda used for the design of the ESBWR Class 1, 2, and 3 piping and components. However, DCD, Tier 2, Table 1.9-22, "Industrial Codes and Standards Applicable to ESBWR," identifies the 2001 edition throughout and including the 2003 addenda of the ASME Code as the code of record. The applicant noted in the table that all limitations and modifications specified in 10 CFR 50.55a must be met. However, the staff notes that the 2001 edition throughout and including the 2003 addenda of the ASME Code is excluded from the seismic design for piping by 10 CFR 50.55a(b)(1)(iii). In Request for Additional Information (RAI) 5.2-75, the staff requested that the applicant specify and document an acceptable ASME Code and ASME Code editions and addenda to be used for the design of ESBWR piping and components, in accordance with the requirements of 10 CFR 50.55a. The staff noted that information regarding the ASME Code of record is a Tier 2* information item, requiring NRC approval if the information must be changed in the DCD, Tier 2 final safety analysis report (FSAR).

The ASME Code is Tier 1 information; however, the specific edition and addenda are Tier 2* information in part because of the continually evolving design and construction practices (including inspection and examination techniques) of the ASME Code. Fixing a specific edition and addenda during the design certification stage might result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although reference to a specific edition of the ASME Code for the design of ASME Code class components and their supports is suitable for reaching a safety finding during the design certification stage, the construction practices and examination methods of an updated ASME Code that would be effective at the combined license (COL) stage must be consistent with the design practices established at the design certification stage.

To avoid this potential inconsistency for the ESBWR pressure-retaining components and their supports, it is appropriate that the ASME Code be specified as Tier 1 information and the specific edition and addenda as Tier 2* information, thereby allowing the COL applicant the option to revise or supplement the referenced ASME Code edition with portions of the later editions and addenda while continuing to ensure consistency between the design and construction practices. This procedure ensures consistency with the latest design, construction, and examination practices. However, the DCD stated that in the case where there is a need to modify certain design parameters associated with a specific ASME Code edition or addenda during its design certification review, particularly if that information is important for establishing a significant aspect of the design or if the staff uses it in reaching its final safety determination. The various sections of the DCD reflect such considerations as necessary. As such, all ASME Code Class 1, 2, and 3 pressure-retaining components and their supports will be designed in accordance with the requirements of Section III of the ASME Code, using the specific edition and addenda given in the DCD. Therefore, the DCD should designate this information as Tier 2* information as it relates to the code edition and addenda.

In its response to RAI 5.2-75, the applicant indicated that it will revise Section 5.2.1.1 of the ESBWR DCD to note the use of the ASME Code, Division 1, Section III, 1992 Edition with 1993 Addenda, for seismic design of piping and the use of ASME Code, 1989 Edition with no addenda, for weld-leg dimensions. In addition, the applicant will also revise DCD, Section 5.2.1.1 to reference the ASME Code of record in Table 1.9-22 as it relates to ASME Code, 2001 Edition throughout and including the 2003 Addenda, which is used for the design of components and supports. The applicant noted that DCD, Tier 2, Table 1.9-22 and Section 5.2.1.1, will include and designate those ASME Code editions and addenda used for the ASME Code, Section III piping and components in the ESBWR as Tier 2* information requiring NRC approval for changes. The staff finds the ASME Code editions and addenda used for the ESBWR design of ASME Code Class 1, 2, and 3 piping and components, to be in compliance with 10 CFR 50.55a. Therefore, they are acceptable. This was tracked as Confirmatory Item 5.2-75. The staff reviewed Revision 6 of the DCD and finds that the above information has been properly incorporated. Therefore, the confirmatory item is closed.

In RAI 3.12-1, the staff requested that the applicant explain how it will satisfy the requirements of 10 CFR 50.55a(b). Section 3.12.3.1 of this report discusses the resolution of this issue.

By letter dated March 12, 2010, the applicant requested the use of Code Case N-782 for ESBWR design. This Code Case is not included in NRC RG 1.84 Revision 34. In accordance with 10 CFR 50. 55a (a)(3), the applicant submitted justification requesting NRC approval for the use of this ASME Code Case as a proposed alternatives to the rules of Section III Subsection NCA-1140 regarding applied code Editions and Addenda, that is required by 10 CFR 50.55a(c), (d) and (e).

Code Case N-782 provides that the Code Edition and Addenda endorsed in a design certified or licensed by the regulatory authority may be used for systems and components constructed to ASME Code, Section III requirements. These alternative requirements are in lieu of requirements that base the Edition and Addenda on the construction permit date. Reference to Code Case N-782 will be included in component and system design specifications and design reports to permit certification of these specifications and reports to the Code Edition and Addenda cited in the DCD and approved by the NRC.

The applicant indicates that the proposed alternative provides an acceptable level of quality and safety, because the NRC endorses the quality and safety of the ASME BPV Code editions and addenda at the time of certification of the design to be at an acceptable level. The use of Code Case N-782 facilitates the use of the ASME Code edition and addenda included in the ESBWR Design Certification. Therefore, it will provide the same level of quality and safety as was included in the information reviewed for the ESBWR Design Certification. The applicant also indicated that Code Case N-782 is needed so that design specifications and reports using the 2001 Edition through the 2003 Addenda of the Code, approved in the Design Certification, can be approved for COL applications. Without NRC approval of Code Case N-782, future Combined License applicants would be required to seek a departure from the certified design. This is a hardship without a compensating increase in the level of quality and safety and could result in a decrease of standardization. The information provided in this letter is generic and applies to all COL applicants referencing the ESBWR design certification.

The staff finds that the applicant has provided adequate justification for use of Code Case N-782 as a proposed alternatives to the rules of Section III Subsection NCA-1140 in accordance with requirements of 10CFR50.55a(a)(3) and concludes that this Code Case N-782 is therefore, acceptable to be used in ESBWR.

5.2.1.1.4 Conclusions

Based on its review, the staff finds that the ESBWR ASME Code of record, including the editions and addenda, as set forth above in Section 5.2.1.1.3, complies with 10 CFR 50.55a and is, therefore, acceptable. As a result, the staff finds that the construction of all ASME Code Class 1, 2, and 3 components and their supports will conform to the appropriate ASME Code editions and addenda, as well as the NRC's regulations, and that component quality will be commensurate with the importance of the safety function of all such components and their supports.

5.2.1.2 Applicable Code Cases

5.2.1.2.1 Regulatory Criteria

The staff reviewed DCD, Tier 2, Revision 7, Section 5.2.1.2, in accordance with SRP Section 5.2.1.2, Revision 3. The staff's acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

- GDC 1, as it relates to the requirement that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed

- 10 CFR 50.55a, as it relates to the rule that establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of BWRs and pressurized-water reactors by requiring conformance with appropriate editions of specified published industry codes and standards

5.2.1.2.2 Summary of Technical Information

DCD Tier 2, Section 5.2.1.2, Revision 7, states that the ESBWR meets the requirements of (1) GDC 1, as it relates to the requirement that safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed, and (2) 10 CFR 50.55a, as it relates to the rule that establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of BWRs by requiring conformance with appropriate editions of specified published industry codes and standards. To meet these requirements, the applicant identified in ESBWR DCD Table 5.2–1 various ASME Code cases that are applicable to the component design, construction, and inspection. The NRC staff has either accepted or conditionally accepted all ASME Code cases identified in Table 5.2-1, as discussed in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," and RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."

5.2.1.2.3 Staff Evaluation

To meet the requirements of GDC 1 and 10 CFR 50.55a, RGs 1.84, 1.147, and 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," issued June 2003, identify the acceptability of ASME Code cases that may be applied in the construction of pressure-retaining ASME Code Class 1, 2, and 3 components. The only ASME Code cases acceptable for use in the design of ASME Code Class 1, 2, and 3, piping systems in the ESBWR are those that are either conditionally or unconditionally approved in RG 1.84 and are in effect at the time of design certification.

DCD, Table 5.2-1 identifies specific ASME Code cases that will be applied in the construction of pressure-retaining Class 1, 2, and 3 components covered by ASME Code, Section III. The staff's review of this table is based on the guidelines in RG 1.84, which discusses the applicable ASME Code cases which the NRC staff has either conditionally or unconditionally endorsed. The staff has endorsed all of the 13 ASME Code cases identified in Table 5.2-1 of the DCD and included in RG 1.84.

In RAI 5.2-32, the staff asked the applicant to provide either annulled ASME Code cases that are not included in DCD, Tier 2, Table 5.2-1, or ASME Code cases that are under development which will potentially be applied in the design and construction of ESBWR pressure-retaining Class 1, 2, and 3 components covered by ASME Code, Section III. In its response, the applicant indicated that it is not planning to use any annulled ASME Code cases that are not already included in DCD, Tier 2, Table 5.2-1. The applicant is not aware of any ASME Code cases under development that would be needed for the RCPB. In RAI 5.2-33, the staff noted that DCD, Tier 2, Table 5.2-1, lists ASME Code Case N-71-17 for the design and construction of the ESBWR, although the current approved revision is ASME Code Case N-71-18. The staff asked the applicant to justify the differences between the two revisions in the ESBWR design application. The applicant responded that it will correct Table 5.2-1 to indicate the application of ASME Code Case N-71-18. However, the applicant subsequently deleted this ASME Code case from Tier 2 in Revision 3 of the DCD. In response to RAI 5.2-34, the applicant indicated that it will use only those ASME Code cases approved in RG 1.84 for the design of ASME Code Class 1, 2, and 3 components in the RCS. The ASME Code cases that pertain to ASME Code, Section XI, Division 1, as approved in RG 1.147, are used only as they relate to preservice and inservice inspection (PSI/ISI) of ASME Code components. In RAI 5.2-50, the NRC requested that the applicant discuss those ASME Code cases listed in Table 5.2-1 which the NRC has not approved for use (i.e., ASME Code Cases N-634 and N-491-2) and include a basis for their use. RAI 5.2-50 was being tracked as an open item in the SER with open items. The applicant subsequently deleted ASME Code Cases N-634 and N-491-2 in Revision 5 of the ESBWR DCD. The staff finds this acceptable because the applicant no longer lists unapproved ASME Code Cases. In response to RAI 5.2-50 S02, the applicant indicated that it will use American Society of Testing and Materials (ASTM) A709 HPS 70W material for containment internal structures. The use of this material falls under ASME Code, Section III, Division 2, and is not applicable to ASME Code cases used for RCPB components. Section 3.8 of this report discusses the ASME Code cases used for ASME Code, Section III, Division 2, applications. RAI 5.2-50 and associated open item are, therefore, resolved.

On the basis of the above evaluation, the staff finds that the applicant will no longer use certain unapproved code cases as they are deleted from DCD, Tier 2, Table 5.2-1. The staff also finds that all of the ASME Code cases listed in DCD Table 5.2-1 meet the guidelines of RG 1.84 in that the staff has reviewed and endorsed these ASME Code cases. The staff finds that the applicant's compliance with the requirements of these ASME Code cases will result in component quality that is commensurate with the importance of the safety functions of the affected components.

5.2.1.2.4 Conclusions

The NRC staff has reviewed the ASME Code cases listed in Table 5.2-1 of ESBWR DCD, Tier 2, which meet the guidelines of RGs 1.84, 1.147, and 1.192. The specified ASME and American National Standards Institute (ANSI) Code cases that will apply in the construction of components covered by ASME Code, Section III, Division 1, Class 1, 2, and 3, and Class MC are consistent with the requirements of 10 CFR 50.55a and GDC 1, as well as the guidance provided in RGs 1.84, 1.147, and 1.192. Therefore, the staff considers that the applicant's compliance with the requirements of these ASME Code cases will result in component quality that is commensurate with the importance of the safety functions of the affected components.

5.2.2 Overpressure Protection

5.2.2.1 Regulatory Criteria

The staff reviewed DCD, Tier 2, Section 5.2.2,Revision 7, in accordance with SRP Section 5.2.2, draft Revision 3, "Overpressure Protection."

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 beyond those identified in the version used by the staff. Therefore, the staff finds that the use of draft Revision 3 of SRP Section 5.2.2, issued in 1996, is acceptable for this review.

During power operation, SRVs, SVs, and the reactor protection system provide overpressure protection for the RCPB. For the ESBWR, the staff's review covered the SRVs and SVs on the MSLs and piping from these valves to the suppression pool and the drywell.

Acceptance criteria are based on GDC 15, "Reactor Coolant System Design," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary." Specifically, the acceptance criteria are based on GDC 15 as it relates to the design of the RCS and associated auxiliary, control, and protection systems having sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation.

In addition, SRP Section 5.2.2 states that the acceptance criteria are based on GDC 31 as it relates to the fracture behavior of the RCPB. Section 5.2.3 of this report addresses this review area. Overpressure protection during low-temperature operation is not considered for the ESBWR, since there is a very low probability of the ESBWR operating in water-solid conditions. Therefore, this report does not address overpressure protection during low-temperature conditions for the ESBWR.

The ESBWR design must meet the requirements of 10 CFR 50.34(f) which reference Three Mile Island-2 (TMI-2) Action Items II.D.1 "Testing Requirements, II.D.3, "Relief and Safety Valve Position Indication," and II.K.3.16, "Reduction of Challenges and Failures of Relief Valve Feasibility Study and System Modification."

5.2.2.2 Summary of Technical Information

DCD, Tier 2, Sections 5.2.2 and 15.5.1, Revision 7, describe the overpressure protection and the overpressure protection analyses.

The NBS relief system consists of 10 SRVs and 8 SVs located on the MSL between the RPV and the inboard MSIV. The SRVs and the SVs provide the two main protection functions of overpressure protection and automatic depressurization.

The SRVs and SVs function as SVs and open by steam pressure to prevent NBS overpressurization. The safety mode of operation is initiated when direct and increasing static inlet steam pressure overcomes the restraining spring and frictional forces acting against the inlet steam pressure at the valve disc. This moves the disc in the opening direction. The condition at which this actuation is initiated corresponds to the set-pressure value stamped on the nameplate of the valves.

The SRVs and SVs meet the requirements of Section III of the ASME Code. The rated capacity of the SRVs is sufficient to prevent a rise in pressure within the RPV to more than 110 percent of the design pressure during anticipated operational occurrences (AOOs).

Of the 18 total valves, 10 are ADS valves and open automatically during a LOCA to depressurize the RV. The depressurization function is accomplished through the use of SRVs and eight squib-actuated DPVs.

Each SRV has one dedicated, independent pneumatic accumulator, which provides the safety-related nitrogen supply for opening the valve.

The SRVs and SVs are flange mounted onto forged outlet fittings located on the top of the MSL piping in the drywell.

The SRVs and SVs are actuated in groups of valves at staggered times by delay timers as the reactor undergoes a relatively slow depressurization. This minimizes reactor level swell during the depressurization, thereby enhancing the passive resupply of coolant by the GDCS.

The use of a combination of SRVs, SVs, and DPVs to accomplish the ADS function improves ADS reliability against common-mode failures. Because the SRVs serve two different purposes, overpressure protection and automatic depressurization, the number of required DPVs is minimized. Using DPVs for the additional depressurization capability needed beyond what the SRVs can provide minimizes the total number of SRVs, SRV discharge lines, and quenchers in the suppression pool. This arrangement also minimizes the need for SRV maintenance and periodic calibration and testing, as well as the potential for simmering.

The ADS automatically actuates on a low RPV water-level signal that persists for a preset time. Two-out-of-four logic is used to activate the SRVs and DPVs. The persistence requirement for the low RPV water-level signal ensures that momentary system perturbations do not actuate the ADS when it is not required. The two-out-of-four logic ensures that a single failure does not cause spurious system actuation, while also ensuring that a single failure cannot prevent initiation. The ADS may also be manually initiated from the main control room (MCR).

5.2.2.3 Staff Evaluation

The staff assessed the design and function of the ESBWR overpressure protection system.

The pressure relief system for the RCPB does the following:

- prevents the pressure in the RCPB from rising beyond 110 percent of the design value

- provides automatic depressurization for breaks in the NBS so that the GDCS can operate to protect the fuel barrier

DCD, Tier 2, Section 5.2.2.1, states, "the rated capacity of the pressure relieving devices shall be sufficient so that the rise in pressure within the protected vessel does not exceed 120 percent of the design pressure for pressurization events described in Chapter 15." Based on a review of Chapter 15, the staff's position is that the pressure limit for pressurization events is 110 percent of the design pressure. Therefore, the applicant should revise the DCD to include 110 percent instead of 120 percent. The staff tracked RAIs 15.0-17 and 15.0-18 as open item in the SER with open items. The applicant revised the DCD Section 5.2.2 to state that the RPV will not exceed 110 percent of the design pressure and the staff verified that this change was incorporated into the DCD; therefore, RAIs 15.0-17 and 15.0-18 and associated open item are resolved.

For the pressure relief system to be acceptable, it must be possible to verify its operability and its ability to withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, and faulted conditions. Section 3.6 of this report evaluates protection against the dynamic effects associated with the postulated rupture of piping.

For overpressure protection, the ICs have sufficient capacity to preclude actuation of the SRVs during normal operational transients. The SRVs limit the pressure to less than the RCPB design pressure during more severe transients.

Ten SRVs and 8 SVs provide overpressure protection in the ESBWR. The nominal pressure setpoint of the 10 SRVs is 8.618 megapascals (MPa) (1,250 pounds per square inch gage (psig)), and the nominal set pressure for the 8 SVs is 8.756 MPa (1,270 psig). The SRVs and SVs are mounted on the four MSLs between the RV and the first isolation valve inside the drywell. Ten ADS SRVs discharge through piping to the suppression pool, and eight non-ADS SVs discharge into the drywell. Short discharge pipes with end-mounted rupture disks limit SV discharge from entering the drywell atmosphere during normal operation. Two vacuum relief valves on each SRV discharge line minimize the initial rise of water in the discharge piping.

ADS SRVs are provided with nitrogen accumulators and check valves. These accumulators ensure that the valves can be opened following the failure of the normal gas supply. The accumulator capacity is sufficient for one actuation at drywell design pressure.

The SRVs and SVs are classified as QG A and seismic Category I, as shown in DCD, Tier 2, Table 3.2-1. The design of the SRVs and SVs is consistent with the guidance in RGs 1.26 and 1.29, Revision 3, "Seismic Design Classification," issued September 1978.

The ADS SRVs can also be operated in the relief mode by remote-manual controls from the MCR.

GDC 15 defines the basis for overpressurization protection in a nuclear reactor. It requires that the RCPB design conditions not be exceeded during any condition of normal operation, including AOOs. To satisfy this criterion, the overpressurization protection system for the ESBWR is designed to comply with ASME Code, Section III, which requires that the maximum pressure reached during the most severe pressure transient be less than 110 percent of the design pressure. For the ESBWR, that pressure limit is 9.48 MPa (1,375 psig). The applicant used the computer simulation model TRACG to analyze a series of transients that would be expected to require SRV actuation to prevent overpressurization. GEH Topical Report, "TRACG Model Description," NEDE-32176P, Revision 4, issued January 2008, describes the TRACG model. Section 21.6 of this report provides the staff's evaluation of the TRACG model.

The staff reviewed the overpressure analyses presented in Section 15.5.1 and found that the applicant's assumptions are consistent with the assumptions given in SRP Section 5.2.2. For the most severe transient (i.e., closure of all MSIVs with a high neutron flux scram), the maximum vessel bottom pressure is calculated to be less than the acceptance limit of 9.48 MPa gage (1,375 psig). The analysis assumed that the plant was operating at a rated steamflow of 2,433 kilograms per second (kg/s) (19.3180 million pounds per hour (Mlbm/hr)) and a vessel dome pressure of 7.17 MPa (1,040 psig). The analysis credits the spring action safety mode of only one valve. The ESBWR RPV is larger than that in the currently operating BWRs, and therefore, the reactor pressurization is slower. In general, RPV pressure ceases to increase once a single relief valve opens because of the higher steam volume-to-power ratio of the ESBWR, which causes the pressure increase rate before a scram to be much lower than currently operating BWRs. After a scram, the pressure increase rates resulting from stored energy release are correspondingly lower.

The applicant based the sizing of the SRVs on the initiation of a reactor scram by the high neutron flux scram, which is the second safety-grade scram signal from the reactor protection system following MSIV closure. The staff notes that the spring action mode of only one valve is required for reactor overpressure protection, however, all of the 18 valves are required for an anticipated transient without scram (ATWS), as described in Section 15.5.4 of this report. The staff believes that the qualification and redundancy of reactor protection system equipment, coupled with the limitation of the RPV to less than 110 percent of design pressure, provide adequate assurance that the RV integrity will be maintained for the limiting transient event.

As required by 10 CFR 50.34(f)(1)(vi), which references TMI-2 Action Item II.K.3.16, a study must be performed to identify practical system modifications that would reduce challenges and failures of relief valves in BWRs, without compromising the performance of the valves or other systems. The SRVs are expected to open in the event of an ATWS or the occurrence of beyond design-basis events. However, one of the key design criteria of the ESBWR is that SRVs shall not need to open during most transients to protect against overpressure. Rather, overpressure protection is achieved through the use of the ICS. GEH and the Boiling Water Reactor Owners Group (BWROG) responded to this requirement for their earlier boiling-water reactor (BWR) models. Based on a review of the existing operating information on the challenge rate of relief valves, BWROG concluded that the BWR/6 product line had already achieved a level of reduction in the SRV challenge rate. The principal reason for this reduction is that the BWR/6 uses direct-acting SRVs, rather than the pilot-operated design used in some earlier BWRs. The ESBWR uses a modern and improved SRV and SV design; therefore, earlier problems are not expected to occur. The NRC staff finds that the ESBWR design complies with 10 CFR 50.34(f)(1)(vi) and TMI-2 Action Item II.K.3.16.

In Revision 2 of the DCD, the applicant deleted the following statement found in DCD, Revision 1, Section 5.2.6, without including a reason for the deletion:

> The COL applicant is required to submit an overpressure protection analysis for core loadings different than the reference ESBWR core loading.

Rather than deleting this sentence, the staff believes that the applicant should have revised it to state the following:

> The COL applicant is required to submit an overpressure protection analysis for the actual core for the initial startup.

The staff tracked RAI 5.2-61 as an open item in the SER with open items. In response to RAI 5.2-61, the applicant stated that the overpressure protection analysis for the initial core is included in NEDC-33337, "ESBWR Initial Core Transient Analyses," issued October 2007. The analysis with only one safety relief valve showed that there is sufficient margin for the reactor overpressure protection. If credit is given to all the ten safety relief valves in the analysis, it is expected that there will be significant margin for overpressure protection. Moreover, the pressurization transients in the ESBWR are expected to be less severe than in current operating BWRs, therefore the staff decided that the applicant need not submit cycle-specific overpressure protection analyses for staff review. Therefore, RAI 5.2-61 and associated open item are resolved.

As required by 10 CFR 50.34(f)(2)(x), which references TMI-2 Action Item II.D.1, licensees must provide a test program with associated model development and conduct tests to qualify RCS relief and SVs for all fluid conditions expected under operating conditions, transients, and accidents. The test program must consider ATWS conditions. For currently operating plants, a generic test program for current valve designs and plant-specific responses for individual plant piping configurations and system responses resolved this issue. The applicant must either confirm that the generic test program for currently operating plants is applicable to ESBWR transients and accidents or commit to perform the required testing and provide necessary plant-specific testing. In ESBWR DCD, Tier 2, Table 1A-1, "TMI Action Plan Items," the applicant states that the SRVs will be tested at a suitable test facility in accordance with quality control procedures to detect defects and to prove operability before installation. The conducted tests will include hydrostatic, steam leakage, full-flow pressure and blowdown, and response time testing. The valves will be installed as received from the factory. The valve manufacturer will certify that design and performance requirements, including capacity and blowdown, have been met. The vendor will adjust, verify, and indicate the setpoints on the valves. Specified manual and automatic initiation signals for power actuation of each ADS SRV will be verified during the preoperational test program described in Chapter 14. The applicant also stated that the inspection and test program for the SRVs will follow a quality assurance program that complies with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. SRV setpoints will not be tested in place, but the SRVs will be removed for maintenance or bench testing and reinstalled during normal plant shutdowns. The valves will be tested to check set pressure in accordance with the requirements of the plant's technical specifications (TS). Further, as discussed in Section 3.9.3 of this report, the external and flange seating surfaces of the SRVs will be visually inspected when the valve is removed for maintenance or bench testing. The staff finds these actions to be consistent with the applicable TMI Action Item II.D.1 provision and to address lessons learned from SRV performance at operating nuclear power plants. Therefore, as discussed in Section 3.9.3 of this report, the staff finds the applicant's response to be acceptable and TMI Action Item II.D.1 is resolved.

In accordance with the requirements of 10 CFR 50.34(f)(2)(xi), which references TMI-2 Action Item II.D.3, the control room includes SRV and SV position indications.

SRV setpoint drift and seat leakage are generic problems. In the response to RAI 5.2-20, the applicant addressed (1) specific design features of the ESBWR SRVs, (2) a comparison of the relative performance of ESBWR SRVs and SRVs currently installed in operating reactors, and (3) a detailed description of any improvements between the ESBWR SRV design and the design of SRVs presently installed in operating reactors in terms of seat leakage, setpoint drift, and actuator reliability.

In the response to RAI 5.2-21, the applicant addressed (1) improvements in the air actuator, especially materials used for components such as diaphragms and seals, (2) safety margins associated with the air accumulator design, (3) pressure indications in the accumulator and how this information is relayed to the operator, and (4) provisions employed to ensure that valve and valve actuator specifications include design requirements for operation under expected environmental conditions (i.e., radiation, temperature, humidity, and vibration).

In the response to RAI 5.2-20, the applicant stated that it had not finalized the detailed design and selection of the ESBWR SRVs. In the response to RAI 5.2-22, the applicant stated that it will prepare a purchase specification for the SRVs which uses the applicant's environmental qualification experience base. The SRVs will be subject to the environmental and dynamic qualification program. In the response to RAI 5.2-7, the applicant stated that, consistent with past practice, it will prepare a purchase specification for the SRVs which addresses the inspection and test requirements of the program. In regard to RAIs 5.2-7, 5.2-20, and 5.2-22, the NRC staff requested that the applicant specify its acceptance criteria for the design and qualification of the SRVs to be used in the ESBWR, including appropriate inspection, test, analysis, and acceptance criteria (ITAAC). The applicant responded that Item 1 in DCD, Tier 1, Revision 3, Table 2.1.2-2, contains an ITAAC to confirm the basic configuration for the NBS and states that those inspections must be conducted using the acceptance criteria that the as-built NBS conforms to the basic configuration, as defined in DCD, Tier 1, Section 2.1.2. The applicant believes that this ITAAC includes programmatic reviews of SRV design and environmental qualifications which meet the intent of the supplemental RAI in which the staff requested that the applicant specify its acceptance criteria for the design and qualification of the ESBWR SRVs, including appropriate ITAAC.

DCD, Tier 1, Revision 3, Section 1.1.12.2.1(4), states that the basic configuration ITAAC includes the following:

> Type tests or type tests and/or analyses, of the safety related mechanical equipment demonstrate qualification to applicable normal, abnormal and design basis accident conditions with out loss of the safety-related function for the time needed during and following the conditions to perform the safety related function considering the applicable harsh environmental conditions.

The NRC staff responded with the following supplemental request:

A. The referenced ITAAC is not sufficient. Revise the ITAAC table to include verification for the SRV discharge capacity and set points to demonstrate that the as-built is consistent with the assumptions of the safety analyses.

B. Include a COL Applicant or COL Holder Item to the DCD to ensure that operating experience, for example, issues identified in Regulatory Issue

Summary 00-012, “Resolution of Generic Safety Issue B-55, ‘Improved Reliability of Target Rock Safety Relief Valves,’” IE Circular 79-18, “Proper Installation of Target Rock Safety Relief Valves,” BL 74-04, “Malfunction of Target Rock Safety Relief Valves,” and NUREG-0763, “Guidelines for Confirmatory In-plant Tests of Safety Relieve Valve Discharges for BWR Plants” are addressed when the SRVs are procured.

C. Revise the DCD, Tier 1, Section 1.2.2.1 to expand the environmental qualification verifications to include mechanical equipment such as seals and gaskets.

The staff tracked RAIs 5.2-20 and 5.2-22 as open item in the SER with open items.

In response to the staff’s concern regarding generic problems with pilot-operated SRVs (e.g., setpoint drift, seat leakage), the applicant stated in its response to RAI 5.2-20 S03, that it will consider operating experience when selecting the design of the SRVs and SVs. The following generic communications will be factored into the selection of SRVs and SVs:

- Generic Safety Issue B-55, “Improved Reliability of Target Rock Safety Relief Valves”
- NUREG-0763, “Guidelines for Confirmatory In-Plant Tests of Safety Relief Valve Discharges for BWR Plants”
- Inspection and Enforcement Office (IE) Circular 79-18, “Proper Installation of Target Safety Relief Valves”
- BL 74-04, “Malfunction of Target Rock Safety Relief Valves”

The applicant indicated that the ESBWR will use either direct-acting SRVs or a modern pilot-operated SRV design that has been proven not to experience the performance problems observed in earlier BWRs. As discussed in Section 3.9.6 of this report, since the applicant agreed to consider the operating experience when selecting the design of the SRVs, the applicant’s response is acceptable.

In response to RAI 5.2-22, the applicant included SRV and SV discharge capacity and setpoints verification in the ITAAC. The staff will verify this during the ITAAC closure phase.

The applicant provided necessary information required by the staff; therefore, RAIs 5.2-20 and 5.2-22 and associated open items are resolved.

Operating experience has shown that SRV failure may be caused by exceeding the manufacturer’s recommended service life for the internals of the SRV or air actuator. In addition to periodic testing, the licensee shall perform valve inspection and overhaul in accordance with the manufacturer’s recommendations. In response to RAI 5.2-25, the applicant stated, “Every 5 years during reactor plant shutdown, the valves are subjected to a complete visual examination, set pressure testing and seat tightness testing.” The licensee will test SRVs in accordance with the inservice testing (IST) program as discussed in Section 3.9.6 of this report.

The effects of flow-induced SRV discharge line back pressure on the performance of the SRV are addressed by sizing the line to ensure that the steady-state back pressure does not exceed 40 percent of the SRV inlet pressure. This sizing criterion controls the effective back-pressure buildup and maintains the required force balance needed to keep the SRV open and permit proper blow down. The non-ADS SRVs discharge through the rupture discs to the drywell. In response to RAI 5.2-10, the applicant stated that the design of the rupture disc will comply with ASME Code, Subsection NB-7623.

Before the valves are installed, the SRV manufacturer will test the valves hydrostatically according to the requirements of ASME Code, Section III. During startup testing, opening response time and set-pressure tests will be conducted to verify that design and performance requirements have been met.

5.2.2.4 Conclusions

For the reasons set forth above, staff finds that the pressure relief system, in conjunction with the ICS and the reactor protection system, will provide adequate protection against overpressurization of the RCPB. The staff further finds that the overpressurization system is acceptable and meets the relevant requirements of GDC 15.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Regulatory Criteria

The staff reviewed DCD, Tier 2, Section 5.2.3, Revision 7 in accordance with SRP Section 5.2.3, Revision 3. The materials specifications, compatibility of materials with the reactor coolant, fabrication and processing of ferritic materials, and fabrication and processing of austenitic stainless steel within the RCPB are acceptable if they meet the relevant requirements set forth in 10 CFR 50.55a; GDC 1, 4, “Environmental and Dynamic Effects Design Bases,” 14, “Reactor Coolant Pressure Boundary,” 30, “Quality of Reactor Coolant Pressure Boundary,” and 31; Appendix B to 10 CFR Part 50; and Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50. These requirements are discussed below:

- Compliance with GDC 1 and 10 CFR 50.55a requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

- Compliance with GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including LOCAs.

- Compliance with GDC 14 requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

- Compliance with GDC 30 requires that components of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical.

- Compliance with GDC 31 requires that the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

- Compliance with Appendix B to 10 CFR Part 50 requires, in Criterion XIII, that measures be established to control the cleaning of material and equipment to prevent damage or deterioration.

- Compliance with Appendix G to 10 CFR Part 50 requires that the fracture toughness of RCPB ferritic materials be tested in accordance with the requirements of the ASME Code and that the pressure-retaining components of the RCPB that are made of ferritic materials meet requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including AOOs.

5.2.3.2 Summary of Technical Information

DCD, Tier 2, Table 5.2-4, "Reactor Coolant Pressure Boundary Materials," lists the principal pressure-retaining materials and material specifications for the RCPB components. This list includes the MSIVs, SRVs and DPVs, main steam piping, CRD components, RPV, IC piping, and FW piping.

The materials used in the RCPB, including materials that do not act as a pressure boundary, consist of austenitic wrought and cast stainless steel, nickel-based alloys, carbon and low-alloy steels, 400 series martensitic stainless steel, Colmonoy and Stellite hard-facing alloys, and precipitation-hardened stainless steels. The applicant indicated that it considered the compatibility of the materials of construction used in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the system is exposed in the ESBWR design. All of the construction materials are resistant to stress-corrosion cracking (SCC) in the BWR environment. General corrosion of all materials, with the exception of carbon and low-alloy steel, is negligible. The applicant considered the extent of the corrosion of ferritic low-alloy steels and carbon steels in contact with the reactor coolant in the design by providing corrosion allowance for all exposed carbon steel and low alloy steel surfaces.

The ESBWR design complies with RG 1.44, "Control of the Use of Sensitized Stainless Steel," issued May 1973; RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," issued February 1973; GL 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," dated January 25, 1988; and NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," issued January 1988.

Fracture toughness for ASME Code Class 1 ferritic materials used for the reactor piping, pumps, and valves meets the impact testing requirements of ASME Code, Subsections NB-2331 and NB-2332. Materials for bolting meet the requirements specified in Subsection NB-2333.

The ESBWR design provides alternatives to the guidelines in RG 1.50, "Control of Preheat Temperature Employed for Welding of Low-Alloy Steel," issued May 1973, and RG 1.71, Revision 1, "Welder Qualification for Areas of Limited Accessibility," issued March 2007.

Wrought tubular products that are used for pressure-retaining components of the RCPB are subject to the examination requirements of ASME Code, Section III, Subsection NB.

These RCPB components meet the requirements of Appendix B to 10 CFR Part 50 and the ASME Code, thus ensuring adequate control of product quality.

5.2.3.3 Staff Evaluation

As discussed below, the staff evaluated material specifications, compatibility of the materials with the reactor coolant, fabrication and processing of ferritic materials and fabrication, and processing of austenitic stainless steel.

5.2.3.3.1 Material Specifications

The specifications for pressure-retaining ferritic materials, nonferrous metals, and austenitic stainless steels, including weld materials that are used for each component in the RCPB must meet the requirements of GDC 1 and 30 and 10 CFR 50.55a, as they relate to quality standards for design, fabrication, erection, and testing. These requirements are met for material specifications by complying with the appropriate provisions of the ASME Code, by applying the ASME Code cases identified in RG 1.84, and by complying with the guidelines of NUREG-0313, Revision 2.

The staff reviewed DCD, Tier 2, Section 5.2.3.1,to determine the suitability of the RCPB materials for this application. The staff determined that the applicant's material specifications listed in DCD, Tier 2, Section 5.2.3 and Table 5.2-4, for the ESBWR design conform with the guidance in RG 1.84 and NUREG-0313, Revision 2, as well as the appropriate provisions of the ASME Code and other staff guidance except as noted below.

Adhering to the guidance provided in NUREG-0313, Revision 2, appropriately addresses GL 81-03, "Implementation of NUREG-0313, Technical Report on Material Selection & Processing GL for BWR Coolant Press Boundary Piping," dated February 26, 1981, and GL 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping." In addition, NUREG-0933, "A Prioritization of Generic Safety Issues," Task Action Plan Issue A-42 related to pipe cracks in BWRs, New Generic Issue 119.4 related to the use of materials resistant to intergranular stress-corrosion cracking (IGSCC), and New Generic Issue 86 related to the long-range plan for dealing with SCC in BWR piping are resolved for the ESBWR design based on the applicant's adherence to the guidance provided in NUREG-0313, Revision 2.

The staff noted that Table 5.2-4 did not include the material specifications and grades of some major components. The table did not include isolation valves and check valves in the ASME Code Class 1 portion of the FW piping. Table 5.2-4 must include the material specifications for these components. The staff tracked this issue as open item (RAI 5.2-36) in the SER with open items. The staff subsequently reviewed the applicant's modifications to Table 5.2-4 in Revision 5 of the DCD. The applicant modified Table 5.2-4 to include material specifications for FW valves in the RCBP. During its review of modifications to Table 5.2-4, the staff identified portions of the table that required clarification and issued RAI 5.2-36 S02, to resolve the issues listed below.

For DPV bodies, FW valves, and FW disc, the applicant listed SA-426, "Specification for Centrifugally Cast Ferritic Alloy Steel Pipe for High Temperature Service," Grade CP22. The staff requested that the applicant explain why it selected a cast pipe specification for valve bodies and valve disc. The staff also noted that the Table 5.2-4 references to SA-376 TP 304L and 316L must be listed as TP 304LN and TP 316LN to be consistent with SA-376 and ASME Code, Section III, Part D.

The applicant listed SFA-5.23 FS8PO-ECF2-F2H2 under welding filler metals for P3, G3 base materials. The staff noted that this classification specifies that the welding flux is made solely from crushed slag or is a blend of crushed slag with unused flux. The staff requested that the applicant explain how a consistent weld metal chemistry will be maintained using crushed slag or a combination of crushed slag and unused flux rather than unused flux alone. In addition, neither of the submerged arc welding specifications listed in Table 5.2-4 contains an "N" designator for special requirements related weld filler metal used in the core beltline. The staff requested in RAI 5.2-36 S02, that the applicant delete Note 4 of Table 5.2-4 which indicates that filler materials listed in Table 5.2-4 are representative and may be changed.

The applicant responded and provided a proposed revision to Table 5.2-4 to address the staff's concerns identified in RAI 5.2-36 S02. The applicant's proposed revision to Table 5.2-4 deletes specification SA-426 Grade CP22 and replaces it with specification SA-217 Grade WC9. SA-217 is an appropriate specification for the fabrication of valves and Grade WC9 is a low-alloy steel which matches the flow-accelerated corrosion resistance of the FW piping material specified by the applicant. This material is also listed in ASME Code, Section II, Part D as an acceptable material for use in Class 1 systems. Therefore, the staff finds this material acceptable.

The applicant's proposed revision to Table 5.2-4 also changes SA-376 TP 304L and 316L to TP304LN and TP316LN which the staff finds acceptable because the applicant's reference to the above material grades is now consistent with SA-376 and ASME Code, Section III, Part D. In addition, the applicant's revised Table 5.2-4 deletes the use of weld filler metal that includes recycled flux. The staff finds this acceptable because the use of recycled flux could adversely affect final weld metal chemistry. The applicant also modified Table 5.2-4 to add Notes 5 and 6 to address special weld filler metal requirements for the core beltline. Note 5 provides a reference to DCD Table 5.3-1 and Section 5.3.1.5. DCD Table 5.3-1 and Section 5.3.1.5 specifies core beltline composition limits that meet or exceed the requirements of SFA-5.23 for weld material classifications that use the "N" designator. DCD Table 5.2-4, Note 6 requires additional impact testing for core belt line materials per SFA-5.23 "N" designation. The staff finds this acceptable because the core beltline weld materials will meet all applicable requirements of SFA 5.23. The staff subsequently reviewed ESBWR DCD, Revision 6, and verified that the applicant made the modifications to the DCD discussed above. RAI 5.2-36 S02 and its associated open item are therefore resolved.

DCD, Section 3E.2.2 listed SA 672 Gr. C70 as a material used in the RCPB. However, this material was not listed in Table 5.2-4. The staff requested, in RAI 5.2-37, that the applicant correct this inconsistency. In the applicant's response to RAI 5.2-37, it indicated that SA-672 Gr. C70 was listed in error. The applicant provided a proposed revision to 3E.2.2 which listed SA-106 Gr. B and SA-333 Gr. 6. The staff noted that SA-106 Gr. B was not listed in Table 5.2-4. In RAI 5.2-37 S01, the staff requested that the applicant correct this inconsistency. The staff tracked RAI 5.2-37as an open item in the SER with open items. In the applicant's response to RAI 5.2-37 S01, it indicated that it does not intend to use material specification

SA-106 and that this material would be deleted form the DCD for use in the RCPB. The staff reviewed DCD, Revision 5 and verified that the applicant removed references to SA-106 for use in the RCPB.RAI 5.2-37 and the associated open item are resolved.

DCD, Tier 2, Table 5.2-4, indicates that the RCPB includes cast austenitic stainless steel (CASS) components. CASS components used in light-water reactors (LWRs) can be susceptible to thermal aging embrittlement. In RAI 5.2-38, the staff asked the applicant to provide the following information for any CASS component that is part of the RCPB: (1) the impact of this aging effect on the integrity of the components, (2) the consideration of the thermal embrittlement mechanism in the design and material selection for RCPB components, (3) the need for inspections to detect this aging effect, and (4) verification that the δ-ferrite content is calculated using Hull's equivalent factors or a method producing an equivalent level of accuracy. In its response, the applicant referenced its response to RAI 4.5-3, for Items 1, 2, and 3 above. The applicant stated that, at the normal operating temperature for all BWRs of 550 degrees Fahrenheit (F), thermal aging of low carbon stainless steel castings with less than 20-percent ferrite is barely measurable. The applicant also stated that these materials have more than 35 years of operating experience with no problems or failures.

The applicant's responses to Items 1, 2, and 3 above meet the staff's expectation that the applicant screening process for the determination of CASS susceptibility to thermal aging embrittlement, for materials listed in Table 5.2-4, is consistent with the NRC staff position documented in a letter from Christopher I. Grimes of the NRC to Douglas J. Walters of the Nuclear Energy Institute, dated May 19, 2000 (Agencywide Documents Access and Management System Accession No. ML003717179). For Item 4, the applicant stated that it intends to use ASTM A800 to determine δ-ferrite content in lieu of Hull's equivalent factors. This is inconsistent with the staff's position that ferrite content be calculated using Hull's equivalent factors as indicated in NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," issued May 1994. For ferrite content above 12 percent, ASTM A800 may produce nonconservative ferrite levels lower than those calculated using Hull's equivalent factors. In response to RAI 6.1-15, the applicant stated that use of a rigorous statistical analysis can demonstrate that the two methods are equally accurate. In RAI 5.2-38 S01, the staff requested that the applicant provide a rigorous statistical analysis showing that the method to calculate ferrite using ASTM A800 and the method using Hull's equivalent factors are equally accurate. The applicant provided its statistical analysis in response to RAI 5.2-38 S01. The staff reviewed it and found it to be unacceptable because it did not show that ASTM A800 and Hull's equivalent factors are equally accurate. The staff tracked RAI 5.2-38 as an open item in the SER with open items. In response to supplemental RAIs associated with this open item, the applicant informed the staff that it would modify DCD Section 5.2.3.4 to state that, for CASS material used as part of the RCPB or RV internals, the percent ferrite is to be calculated using Hull's equivalent factors, as indicated in NUREG/CR-4513, Revision 1. The applicant also stated that it would modify DCD Section 5.2.3.4 to limit the percent ferrite in CASS material to a maximum value of 20 percent, which is consistent with the NRC staff position regarding the control of thermal embrittlement in CASS materials. The staff reviewed DCD, Revision 5, and verified that the applicant had completed the aforementioned DCD modifications. RAI 5.2-38 and the associated open item are resolved.

Several operating experience issues have arisen related to the fabrication quality and inservice performance of dissimilar metal welds (DMWs) in LWRs. In RAI 5.2-40, the staff asked the applicant to describe DMWs in the RCPB and discuss the selection of filler metals, welding processes, and process controls for DMWs in the ESBWR design.

In its response, the applicant indicated that DMWs are primarily used in the RCPB to join carbon steel to stainless steel piping components. These joints are generally made by applying a buttering layer or layers of 309L or 309MoL followed by completion of the groove weld using 308L, 316L, 309L, or 309MoL. Ferrite content in welds is controlled to between 8FN and 20FN. Postweld heat treatment of the carbon steel after buttering is performed if dictated by ASME Code, Subsection NB-4600. DMWs may also be made using Alloy 82 when welding nickel alloys to carbon steel, low-alloy steel, and stainless steel. Postweld heat treatment of stainless steel components will not be allowed. The staff notes that the aforementioned weld filler materials are considered Category A materials, in accordance with NUREG-0313, and provide an increased level of resistance to IGSCC when compared to non-low-carbon stainless steel welding filler materials. The applicant stated that all of the aforementioned alloys are currently in BWR service with no observed incidences of SCC or other problems which provides additional assurance that the welds will maintain structural integrity throughout the design life of the plant. Based on the above, the staff finds that the applicant's proposed welding methods and selection of weld filler materials for DMWs are acceptable, because the applicant will follow current industry practice; the weld filler materials are consistent with staff guidance and these materials have had favorable operating experience.

ASME Code, Section III, Subsection NB-3121, requires that material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects must provide for these effects during the design or specified life of the component by a suitable increase in or addition to the thickness of the base metal over that determined by the design formulas. In DCD, Tier 2, Section 5.2.3.2, the applicant indicated that it considered the extent of the corrosion of ferritic low-alloy steels and carbon steels in contact with the reactor coolant in the design. In RAI 5.2 43, the staff asked the applicant to describe corrosion allowances for all unclad low-alloy and carbon steel surfaces in the RCPB. In its response, the applicant stated that corrosion allowances for unclad carbon and low-alloy steels are defined for both external (air exposure) and internal (wetted) surfaces. The 60-year allowance for external surfaces is 0.8 millimeters (mm), and the allowance for internal surfaces is 1.6 mm. In response to RAI 6.1-7, the applicant provided additional information regarding its process for determining the corrosion allowance for RCPB ferritic materials. The corrosion allowance is primarily based on the applicant's internal testing. The allowances consider fluid velocity, oxygen content, and temperature and include a safety margin over the actual measured corrosion rates of approximately a factor of 2. The same method, with corresponding allowances, has been applied to most operating BWRs of GEH design, including the advanced boiling-water reactor (ABWR) design. The staff finds this acceptable, given that the applicant has considered the effect of corrosion, based on laboratory testing and operational experience, over the design life of the plant as required by ASME Code, Section III.

DCD, Tier 2, Table 5.2-4, indicates that E9018-B3L and ER90S-B3L will be used to weld components in the RCPB. The staff notes that ASME discontinued the aforementioned weld filler metal classifications several years ago and replaced them with classifications E8018-B3L and ER80S-B3L. The same issue exists in the applicant's proposed revision of Table 6.1-1. In RAI 6.1-2 S02, the staff requested that the applicant modify Tables 5.2-4 and 6.1-1 to include the correct weld filler material classifications. Table 5.2-4 and the applicant's proposed version of Table 6.1-1 list the weld filler material that will be used to weld P5C, G1 materials. After reviewing the RCPB and engineered safety feature (ESF) material specifications provided in the

DCD and the applicant's response, the staff is unable to identify any materials that fall into the P5C, G1 category, in accordance with ASME Code, Section IX, Table QW/QB-422. In RAI 6.1-2 S02, the staff requested that the applicant identify the P5C, G1 materials used in the ESBWR design for RCPB and ESF components or delete this information from the DCD if it does not apply.

Table 5.2-4 and the applicant's proposed revision to Table 6.1-1 identify shielded manual arc welding filler material E8018-G for use in welding low-alloy steel in the ESBWR design. To complete its review and evaluate the applicant's compliance with 10 CFR 50.55a, the staff requested, in RAI 6.1-2 S02, that the applicant provide the complete GEH specification that will be used to purchase E8018-G for the fabrication of ASME Code Class 1, 2, and 3 components. In addition, the staff asked that the applicant provide a technical justification for using the GEH specification in lieu of commercially available welding electrodes. The staff tracked RAI 6.1-2 as an open item in the SER with open items.

The applicant responded and indicated that it would modify Tables 6.1-1 and 5.2-4 to delete obsolete filler material classifications, delete references to P5C, G1 materials, and delete E8018-G filler material classifications. The staff reviewed Revision 5 to the ESBWR DCD and verified that the applicant had made the appropriate modifications. RAI 6.1-2 and the associated open item are resolved.

The staff finds that the applicant's selection of materials for use in the RCPB meets the requirements of the ASME Code or the guidance of RG 1.84 and complies with the guidelines of NUREG-0313, Revision 2, and is therefore acceptable.

5.2.3.3.2 Compatibility of Materials with the Reactor Coolant

The RCPB materials of construction that are in contact with the reactor coolant, contaminants, or radiolytic products must be compatible and must meet the requirements of GDC 4, as they relate to the compatibility of components with environmental conditions. The applicant stated that it considered the compatibility of the materials of construction used in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the RCPB is exposed.

The applicant stated that the materials of construction are compatible with primary coolant water, which is chemically controlled in accordance with the appropriate TS, as discussed in Section 5.4.8 of this report. The applicant's selection of materials and control of water chemistry will ensure compatibility. Additionally, extensive testing and satisfactory performance of these materials in operating plants for several years have proven this compatibility. The materials meet the requirements of GDC 4 because the ESBWR design complies with the applicable provisions of the ASME Code, adheres to the guidance provided in RG 1.44, and conforms to the staff positions of GL 88-01, which are based on the technical information and recommendations provided in NUREG-0313. Therefore, material compatibility with primary water coolant will be assured.

5.2.3.3.3 Fabrication and Processing of Ferritic Materials

The fracture toughness of ferritic materials in the RCPB must meet the requirements of Appendix G to 10 CFR Part 50. These criteria satisfy the requirements of GDC 14 and 31 regarding prevention of fracture of the RCPB.

Appendix G to 10 CFR Part 50 requires the pressure-retaining components of the RCPB to be made of ferritic materials to meet the requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including AOOs. For piping, pumps, and valves, this requirement is met through compliance with the requirements of ASME Code, Section III, Subsection NB-2331 or Subsection NB-2332, and the C_v values specified in Table NB-2332(a)-1. Materials for bolting must meet the impact test requirements of ASME Code, Section III, Subsection NB-2333. Calibration of temperature instruments and C_v impact test machines must meet the requirements of ASME Code, Section III, Subsection NB-2360. The staff reviewed DCD, Tier 2, Section 5.2.3.3.1, and verified that the ESBWR design meets the aforementioned requirements regarding fracture toughness of RCPB piping, components, and bolting and equipment calibration. Section 5.3 of this report presents the staff's evaluation of the fracture toughness requirements of the RPV.

Control of ferritic steel welding by following NRC RGs and adhering to the ASME Code satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a. Adherence to the guidance provided in RG 1.50; RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," issued May 1973; RG 1.34, "Control of Electroslag Weld Properties," issued December 1972; RG 1.71; and ASME Code nonmandatory Appendix D, "Non-mandatory Preheat Procedures," Subsection D-1210, satisfies the aforementioned quality standard requirements.

DCD, Tier 2, Section 5.2.3.3.2, Revision 7 discusses the use of RG 1.50 and preheat requirements when welding low-alloy steel in the ESBWR design. Low-alloy steel is used only in the RPV and FW piping. The applicant indicated that an alternative to RG 1.50 may be applied to the RCPB components. RG 1.50 requires that all low-alloy steel welds be maintained at the minimum preheat temperature until postweld heat treatment is performed. In RAI 5.2-44, the staff asked the applicant to describe the portions of RG 1.50 that will not be followed and the steps that it will take to ensure that delayed cracking of the weld metal or weld heat-affected zone (HAZ) will not occur. The applicant responded that in some cases the RV will be allowed to cool to ambient temperature after application of postweld baking to remove any hydrogen that may be present. Previous BWR licensing documents, including the ABWR final SER, have included this same allowance and it has been accepted by the staff. The applicant indicated that specific postweld baking parameters are dictated by the type of weld involved, the welding process (e.g., inert gas shielded), and prior qualification testing. For example, drop of preheat is allowed for narrow gap, gas tungsten arc welding (GTAW), or gas metal arc welding (GMAW) joints when the weld is subjected to postweld baking for 2 hours at 300 degrees Celsius (C) or 4 hours at 200 degrees C. With gas-shielded welding, there is little potential for the introduction of hydrogen into the weld zone in any case. In accordance with RG 1.50, all such welds will be subjected to volumetric examination to confirm the absence of delayed cracking. All such joints will subsequently receive postweld heat treatment. Therefore, the applicant contends that a combination of postweld baking and inspection meets the intent of the RG. The applicant stated that this process has been successfully applied to operating BWR RVs.

The staff considers the applicant's procedure to perform postweld baking at the temperatures and times stated above for the referenced welding processes to fabricate RCPB components to be an acceptable alternative to the guidance in RG 1.50 which requires the maintenance of preheat until postweld heat treatment is performed. In response to RAI 6.1-4, which references RAI 5.2-44, the applicant indicated that welding processes, such as flux-shielded welding, will require rigorous qualification of the effectiveness of the postweld baking.

The staff notes that this method has been successfully used in several other applications, such as fossil fuel electric generation facilities, as well as petrochemical facilities, with materials that are much more sensitive to hydrogen cracking than those materials used within the RCPB of a nuclear power plant. Postweld baking is an effective measure to prevent delayed hydrogen cracking in welds that do not go directly from preheat temperature to postweld heat treatment. The staff therefore considers the applicant's alternative to RG 1.50 acceptable, given that it provides reasonable assurance that delayed hydrogen cracking will not occur in the time that a weld is completed through completion of postweld heat treatment.

Although the staff finds the applicant's alternative to RG 1.50 acceptable, it requested, in a supplement to RAI 10.3-4, that the applicant modify the DCD to include its alternative to RG 1.50 as it applies to all ASME Code Class 1, 2, and 3 piping and components. In addition, the staff asked that the applicant modify the DCD to include its response to RAI 6.1-4 in which it states that it will apply the minimum preheat recommendations found in ASME Code, Section III, Appendix D, Article D-1000, to all Class 1, 2, and 3 carbon steel and low-alloy steel piping and components in the ESBWR design. The staff tracked RAI 10.3-4 as an open item in the SER with open items.

The staff reviewed Revision 5 of the ESBWR DCD and verified that the applicant appropriately referenced ASME Code, Section III, Appendix D, Article D-1000, and RG 1.50 in DCD Section 5.2.3.3.2. The staff finds this reference acceptable because it meets the acceptance criteria of SRP Section 5.2.3. RAI 10.3-4 and the associated open item, as it applies to the RCPB, are resolved.

DCD, Tier 2, Section 5.2.3.3.2, states that electroslag welding is not allowed on structural weld joints of low-alloy steel. Therefore, RG 1.34 does not apply to the ESBWR design. RG 1.43 applies to clad low-alloy steel, and the DCD identifies the RPV as the only stainless steel clad low-alloy steel component in the RCPB. Section 5.3 of this report evaluates the applicant's adherence to the guidance in RG 1.43 related to RPV fabrication.

In DCD, Tier 2, Section 5.2.3.4.2, the applicant stated that the ESBWR design meets the intent of RG 1.71. In RAI 5.2-45, the staff asked the applicant to discuss its deviations from specific portions of RG 1.71 and explain how those deviations meet the intent of the RG. In its response, the applicant stated that restricted access qualifications are required when access to a nonvolumetrically examined production weld is less than 305 mm (about 12 inches (in.)) in any direction and allows welding from one access direction only. Requalification is required if the production weld is more restricted than the welder's performance qualification. The applicant provided the following rationale:

> If a RCPB weld is subject to volumetric inspection, the inspection method and acceptance criteria will be according to ASME Section III, Subsection NB. If the weld passes this inspection, the weld quality is considered acceptable irrespective of the access restriction. Therefore, the intent of the RG is met by inspection. The fabricator or installer must produce welds that satisfy the Code irrespective of any access restrictions.
>
> The RG indicates restrictions of 12 to 14 inches. Since this is insufficiently definitive from a specification and quality assurance point of view, the applicant selected 12 inches as the defined limit.

> Practically, even though a restriction may exist in one direction from the weld, this is not necessarily the only direction from which the welder may approach the weld. Therefore, if the welder can freely approach the weld from another direction with no access restrictions, the restricted access performance qualification is not required. It is further noted that in the ESBWR design, there are few, if any, RCPB welds that truly have restricted access. Additionally, much of the welding is performed with mechanized welding systems where physical access for a welder is not relevant to the ultimate weld quality.

The staff reviewed the applicant's alternative to RG 1.71 as stated in DCD, Tier 2, Section 5.2.3.4.2. The staff has determined that the applicant's alternative is consistent with the intent of RG 1.71. The applicant's alternative will provide reasonable assurance that welders working in restricted access positions will be appropriately qualified and thus produce sound welds.

For nondestructive examination (NDE) of ferritic steel and austenitic stainless steel tubular products, compliance with the applicable provisions of the ASME Code meets the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards. Subsections NB-2550 through NB-2570, which are discussed in Section 5.2.3.5 of this report, are the applicable provisions of ASME Code, Section III.

5.2.3.3.4 Fabrication and Processing of Austenitic Stainless Steel

All stages of component manufacturing and reactor construction must include process control techniques, in accordance with the requirements of GDC 1, as it relates to nondestructive testing (i.e., examination) to quality standards; GDC 4; and Criterion XIII, "Handling, Storing, and Shipping," of Appendix B to 10 CFR Part 50. These requirements prevent severe sensitization of the material by minimizing exposure of stainless steel to contaminants that could lead to SCC and reduce the likelihood of component degradation or failure through contaminants.

The applicant meets the requirements of GDC 4 and Criterion XIII of Appendix B to 10 CFR Part 50 by complying with the applicable provisions of the ASME Code and following the guidance found in the regulatory positions of RG 1.31, Revision 3, "Control of Ferrite Content in Stainless Steel Weld Metal," issued April 1978; RG 1.36; RG 1.37, Revision 1, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," issued March 2007; RG 1.44; and RG 1.71.

The staff reviewed DCD, Tier 2, Section 5.2.3.4,to ensure that austenitic stainless steel RCPB components are (1) compatible with environmental conditions to avoid sensitization and SCC, (2) compatible with thermal insulation, (3) have appropriate controls on welding and material preservation, and (4) receive appropriate NDE. For NDE of ferritic steel and austenitic stainless steel tubular products, the applicant complied with the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards by specifying the appropriate provisions of the ASME Code, which are in Section III, Subsections NB-2550 through NB-2570. DCD, Tier 2, Section 5.2.3.3.3, states that seamless tubular products must be examined according to ASME Code, Section III, Subsection NB-2550, welded tubular products according to Subsection NB-2560, and cast tubular products according to Subsection NB-2570.

The DCD indicates that all austenitic stainless steels are supplied in the solution heat-treated condition, and special sensitization tests are applied to confirm and ensure proper heat treatment. In RAI 5.2-48, the staff asked the applicant to describe its "special sensitization test" that will be applied to ensure proper heat treatment. In its response, the applicant indicated that the test used to detect susceptibility to intergranular attack is a modified version of ASTM A262, Practice A, wherein rejectable ditching is defined more strictly than in the ASTM version, and retest and acceptance by Practice E is not allowed. The staff considers limiting retest and acceptance by Practice E to be a conservative practice, which, therefore, meets the intent of RG 1.44.

In RAI 5.2-49, the staff asked the applicant to discuss its solution heat treatment requirements for austenitic stainless steel components and welds. In its response, the applicant explained its heat treatment requirements and stated that its solution heat treatment practice is consistent with that described in RG 1.44 and NUREG-0313. The staff finds the applicant's responses to RAIs 5.2-48 and 5.2-49 acceptable because they conform with the guidance provided in RG 1.44 and NUREG-0313.

The ESBWR conforms to the guidance provided in RGs 1.31, 1.44, and 1.36.

The applicant's acceptance criteria for cleaning and cleanliness controls meet the intent of RG 1.37, Revision 1. The applicant provided an alternative to RG 1.37 that is acceptable to the staff. For a discussion refer to Section 4.5.1.2.5 of this report.

5.2.3.4 Conclusions

For the reasons set forth above, the staff finds that the design of the RCPB materials is acceptable and meets the requirements of GDC 1, 4, 14, 30, and 31; Appendices B and G to 10 CFR Part 50; and 10 CFR 50.55a.

5.2.4 RCS Pressure Boundary Inservice Inspection and Testing

5.2.4.1 Regulatory Criteria

The staff reviewed DCD, Tier 2, Section 5.2.4, Revision7, in accordance with SRP Section 5.2.4, Revision 3.

GDC 32, "Inspection of Reactor Coolant Pressure Boundary," requires the periodic inspection and testing of the RCPB, and specific requirements are outlined in 10 CFR 50.55a and detailed in ASME Code, Section XI. Compliance with the preservice and inservice examinations required by 10 CFR 50.55a, as detailed in ASME Code, Section XI, partially satisfies the requirements of GDC 32, as discussed below:

- Compliance with GDC 32 requires, in part, that all components that are part of the RCPB be designed to permit periodic inspection and testing of important areas and features to assess structural and leaktight integrity. Meeting the requirements of GDC 32 ensures an effective periodic inspection program for the RCPB to identify aging effects or other incipient degradation phenomena, thus enabling licensees to take prompt preventive measures to preclude potential loss of coolant or impaired reactor core cooling.

- Compliance with 10 CFR 50.55a requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function they are intended to perform. By reference, 10 CFR 50.55a incorporates Section XI of the ASME Code.

5.2.4.2 Summary of Technical Information

DCD, Tier 2, Section 5.2.4, Revision 7, describes the PSI/ISI and system pressure test programs for NRC QG A, ASME Code Class 1 items. This section describes these programs' implementation of the requirements of Subsection IWB of ASME Code, Section XI. The design to perform preservice inspection (PSI) is based on the requirements of ASME Code, Section XI, 2001 Edition through 2003 Addenda, as specified in DCD, Tier 2, Table 1.9-22. The applicant indicated that the COL licensee is responsible for the development of the PSI/ISI program plans which must be based on the edition and addenda of ASME Code, Section XI, specified in DCD, Tier 2, Table 1.9-22. The ASME Code requirements discussed in this section are provided for information.

5.2.4.3 Staff Evaluation

In DCD, Tier 2, Section 5.2.4, Revision 3, the applicant stated that the development of the PSI/ISI program plans is the responsibility of the COL holder and must be based on ASME Code, Section XI, 2001 Edition through 2003 Addenda. DCD, Tier 2, Section 6.6, Revision 3, indicated that the development of the ASME Code Class 2 and 3 PSI/ISI program plans will be the responsibility of the COL holder and must based on the edition and addenda of ASME Code, Section XI, specified in 10 CFR 50.55a. DCD, Tier 2, Section 6.6, Revision 3, also stated that the COL holder shall specify the edition of the ASME Code to be used, based on the date of issuance of the construction permit or license, in accordance with 10 CFR 50.55a. There appeared to be an inconsistency in the DCD between the editions and addenda of ASME Code, Section XI, which COL applicants are expected to use to develop their PSI/ISI programs.

In RAI 5.2-63, the staff requested that the applicant revise DCD, Tier 2, Sections 5.2.4 and 6.6, to clearly and accurately state the requirements governing the applicable ASME Code edition and addenda to be used by the COL applicant to develop PSI/ISI programs. The staff tracked RAI 5.2-63 as an open item in the SER with open items.

The applicant responded indicating that it would modify DCD, Tier 2, Sections 5.2.4 and 6.6, to state that the ESBWR is designed for the performance of PSI/ISIs, including consideration of the requirements of the ASME Code, Section XI, edition and addenda specified in Table 1.9-22. The applicant further stated that the development of the PSI/ISI programs is the responsibility of the COL holder and will be based on the ASME Code, Section XI, edition and addenda approved in 10 CFR 50.55a(b) 12 months before initial fuel load. The staff finds this acceptable. The staff verified that the applicant made the above modifications to DCD, Tier 2, Sections 5.2.4 and 6.6, Revision 4. RAI 5.2-63 and the associated open item are resolved. Subsequently, in DCD Revision 7, the applicant modified the DCD to state that the licensee will be responsible for the actual development of the ISI/PSI programs. The staff finds this acceptable because licensee is a more appropriate term than COL holder.

5.2.4.3.1 System Boundary Subject to Inspection

The applicant's definition of the RCPB is acceptable if it includes all pressure vessels, piping, pumps, and valves that are part of the RCS, or connected to the RCS, up to and including the following:

- the outermost containment isolation valve in system piping that penetrates the primary reactor containment

- the second of two valves typically closed during normal reactor operation in system piping that does not penetrate primary reactor containment

- the RCS SRVs

The applicant stated, in DCD, Tier 2, Section 5.2.4.1, that the Class 1 system boundary for both the PSI/ISI programs and the system pressure test program includes all of those items within the Class 1 and QG A boundary on the piping and instrumentation schematics. The applicant indicated that based on 10 CFR Part 50 and RG 1.26 the boundary includes the following:

- RPV
- portions of the main steam system
- portions of the feedwater system (FWS)
- portions of the standby liquid control system (SLCS)
- portions of the RWCU/SDC system
- portions of the ICS
- portions of the GDCS

The staff reviewed the information provided in DCD, Tier 2, Section 5.2.4.1, and determined that the ASME Code Class 1 boundary requirements identified by the licensee are consistent with the acceptance criteria in SRP Section 5.2.4.II.1. Section 3.2 of this report presents a detailed staff review of the applicant's classification of ASME Code Class 1 components and piping.

5.2.4.3.2 Accessibility

The design and arrangement of system components are acceptable if adequate clearance is provided in accordance with ASME Code, Section XI, Subsection IWA-1500, "Accessibility."

In DCD, Tier 2, Section 5.2.4.2, the applicant stated that all items within the Class 1 boundary are designed to provide access for the examinations required by ASME Code, Section XI, Subsection IWB-2500. The applicant also stated that ASME Code, Section XI, Subsection IWA-1500 defines considerations for accessibility.

DCD, Tier 2, Section 5.2.4.2, states, under piping, pumps, valves, and supports, that welds are located to permit ultrasonic examination from at least one side, but where component geometries permit, access from both sides is provided. This is acceptable to the staff for ferritic welds, because a one-sided ultrasonic examination can be performed on ferritic materials. However, one-sided ultrasonic examinations cannot be performed on austenitic or DMWs using current technology. For austenitic and DMWs that are accessible from one side only, radiography would be required to attain 100-percent weld coverage for examinations required by ASME Code, Section XI. The staff is concerned that operational experience shows that radiography is not practical in some applications in current operating plants. Difficulty in draining systems and radiological concerns sometimes preclude the use of radiography, resulting in licensees requesting relief from inspection requirements. Designing a system in a manner that will require radiography must include considerations related to operating conditions and radiological concerns to ensure that ISI inspections will be practical to be performed after the plant goes into operation in order to meet the requirements of 10 CFR 50.55a(g). As discussed below, the staff issued several RAIs to address these concerns.

The staff issued several RAIs (6.6-1, 6.6-2, 6.6-3, 6.6-4, 5.2-51, 5.2-53, 5.2-54, 5.2-57, and 5.2-58) regarding the accessibility of components for inspections required by ASME Code, Section XI, and 10 CFR 50.55a. The staff developed RAI 5.2-62, which superseded the aforementioned RAIs, regarding the accessibility and inspectability of welds and components. In RAI 5.2-62, the staff requested that the applicant modify the DCD to (1) specify the inspection methods that are practical to use for inservice inspection (ISI) of welds in ASME Code Class 1 and 2 austenitic and DMWs and (2) add COL applicant items to Sections 5.2.4 and 6.6 to ensure that a COL applicant referencing the ESBWR will provide a detailed description of its plans to incorporate, during design and construction, access to piping systems to enable NDE of such welds during ISI.

ASME Code, Section XI, as incorporated into 10 CFR 50.55a(g), currently allows for either ultrasonic or radiographic examination of welds in ASME Code Class 1 and 2 piping systems. The staff asked that the applicant modify DCD, Tier 1 to state that one or both of these types of examinations are practical for ISI of austenitic and DMWs. The staff notes that ultrasonic examination has advantages with respect to keeping exposures as low as reasonably achievable. With this change to the DCD, any design certification rule that might be issued for the ESBWR will preclude the granting of relief under 10 CFR 50.55a(g)(6) for ISI of such welds. The staff requested that the applicant confirm that austenitic or DMWs in Class 1 and 2 piping systems will be accessible for examination by either ultrasonic or radiographic examination, thus satisfying the requirements of 10 CFR 50.55a(g)(3).

In support of these DCD changes, a COL applicant referencing the ESBWR design certification application should inform the staff of how it plans to meet all access requirements during construction and operation, as required by 10 CFR 50.55a(g)(3)(i) and (ii). The staff notes that the PSI requirements are known at the time a component is ordered, and 10 CFR 50.55a(g) does not provide for consideration of relief requests for impractical examination during the construction phases of the component. The COL items requested above should reflect these considerations. The staff tracked RAI 5.2-62 as an open item in the SER with open items.

The applicant modified DCD, Sections 5.2.4 and 6.6 to describe its design process to ensure that the accessibility of austenitic and DMWs enable the performance of ultrasonic testing or radiographic testing. The staff reviewed the applicant's RAI response and modifications in ESBWR DCD, Revision 5, Sections 5.2.4 and 6.6, and found them to be unacceptable because they did not address a design for accessibility which took into account operational and radiological concerns. The staff issued RAI 5.2-62 S01, and requested that the applicant address this issue.

The applicant modified DCD Sections 5.2.4 and 6.6 to address the staff's concerns. Section 6.6 of this report addresses the accessibility of ASME Code Class 2 components. The applicant proposed to modify DCD Section 5.2.4.2 and include Tier 2* information in lieu of the Tier 1 changes requested by the staff. Given that the COL applicant cannot depart from Tier 2* information without NRC approval, the staff finds that the applicant's proposed modifications described below are acceptable:

> [The ESBWR design includes specific access requirements, in accordance with 10 CFR 50.55a(g)(3), to support preferred UT or optional RT examinations. The design of each component and system takes into account the NDE method, UT or RT, that will be used to fulfill PSI and inservice inspection examination and will take into full consideration the operational and radiological concerns associated with the method selected to ensure that the performance of the required examination will be practical during commercial operation of the plant. Additionally, the design procedural requirements for the 3D layout of the plant include acceptance criteria regarding access for inspection equipment and personnel]*. However, with respect to any design activities for components that are not included in the referenced ESBWR certified design, it is the responsibility of the COL applicant to preserve accessibility to piping systems to enable NDE of ASME Code Class 1 austenitic and DM welds during inservice inspection (COL item 5.2-3-A).

The staff finds that the proposed modifications to DCD Sections 5.2.4.2 and 5.2.6, discussed above, provide assurance that austenitic and DMWs will be accessible so that inspections required by the ASME Code may be performed, taking into account operational and radiological concerns that could affect the practicality of the inspection method chosen for PSI/ISIs. The staff reviewed ESBWR DCD, Revision 6, and verified that the applicant made the modifications to the DCD discussed above. RAI 5.2-62 and its associated open item are therefore resolved.

DCD, Tier 2, Section 5.2.4.2, references a formula, L=2T+152 mm, that is used in the ESBWR design to determine the minimum length for spool pieces. In RAI 5.2-52, the staff asked the applicant to explain how it determined that the distance derived from the formula is adequate for the ESBWR design. In a letter dated August 28, 2006, the applicant stated that the basis for its minimum spool piece length of L=2T+152 mm is ASME Code, Section XI, Appendix D, which specifies L=2T+ 50.8 mm. The 50.8-mm allowance accommodates the transducer footprint. The 2T (T=thickness) distance allows for a full ultrasonic V-path for a 45-degree transducer. The additional 101.2 mm is an allowance for scanner tracks, other beam paths, and the like. The staff finds this acceptable because the applicant has considered the necessary spool piece lengths to facilitate ultrasonic testing examinations.

DCD, Tier 2, Section 5.2.4.2, indicates that items such as nozzle-to-vessel welds often may have inherent access restrictions when vessel internals are installed. Therefore, preservice examination must be performed as necessary to achieve the required examination volume on these items before installation of internals, which would interfere with examination. Section 5.2.4.2 further states that access is sufficient for the inservice examination of the volume described in ASME Code Case N-613-1. The staff finds this acceptable, given that the PSI will be performed in accordance with ASME Code, Section XI, and the ISI of these components will be performed in accordance with ASME Code Case N-613-1, which the NRC endorses in RG 1.147, Revision 14. Use of NRC-endorsed ASME Code cases is permitted by 10 CFR 50.55(g)(3)(i).

The staff reviewed DCD, Tier 2, Section 5.2.4.2, regarding the accessibility of RPV welds, RPV head, RPV studs, and RPV washers. The applicant has incorporated access for examinations of these components into the design of the RPV, biological shield, and vessel insulation to enable the appropriate ultrasonic and visual examinations to be conducted. This includes not only access for remotely operated ultrasonic examination devices, but also sufficient access to perform visual examination during system leakage and hydrostatic testing.

5.2.4.3.3 Examination Categories and Methods

The examination categories and methods specified in the DCD are acceptable if they are consistent with the criteria in ASME Code, Section XI, Subsection IWB-2000, "Examination and Inspection." Every area subject to examination should fall within one or more of the examination categories in Subsection IWB-2000 and must be examined, at least to the extent specified. The requirements of Subsection IWB-2000 also identify the methods of examination for the components and parts of the pressure-retaining boundary.

The applicant's examination techniques and procedures used for PSI or ISI of the system are acceptable, if they conform to the following criteria:

- The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with Subsection IWA-2000 and Subsection IWB-2000 of ASME Code, Section XI.

- The methods, procedures, and requirements regarding qualification of NDE personnel are in accordance with Subsection IWA-2300, "Qualification of Nondestructive Examination Personnel."

- The methods, procedures, and requirements regarding qualification of personnel performing ultrasonic examination reflect the requirements provided in Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," to Division 1 of ASME Code, Section XI. In addition, the performance demonstration for ultrasonic examination systems reflects the requirements provided in Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Division 1 of ASME Code, Section XI.

The staff reviewed DCD, Tier 2, Sections 5.2.4.3.1, "Examination Categories," and 5.2.4.3.2, "Examination Methods," which discuss examination techniques, categories, and methods. The visual, surface, and volumetric examination techniques and procedures conform to the requirements of Subsection IWA-2200 and Table IWB-2500-1 of ASME Code, Section XI, and are therefore acceptable to the staff.

The ASME Code requirements discussed in Section 5.2.4 of the DCD are based on the 2001 edition of the ASME Code, Section XI, with the 2003 addenda. This edition and addenda of Section XI of the ASME Code requires the implementation of Appendix VII for qualification of NDE personnel for ultrasonic examination and the implementation of Appendix VIII for performance demonstration for ultrasonic examination of RCPB piping and components identified in Table IWB-2500. The DCD indicates that ultrasonic examination systems must be qualified in accordance with industry-accepted programs for implementation of the ASME Code, Section XI, Appendix VIII. The staff finds this acceptable.

5.2.4.3.4 Inspection Intervals

The required examinations and pressure tests must be completed during each 10-year interval of service, hereafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of Subsection IWA-2000 concerning inspection intervals of ASME Code, Section XI.

DCD, Tier 2, Section 5.2.4.4, discusses inspection intervals. Subsections IWA-2400 and IWB-2400 of ASME Code, Section XI, define inspection intervals. The inspection intervals specified for the ESBWR components are consistent with the definitions in Section XI of the ASME Code and, therefore, are acceptable.

5.2.4.3.5 Evaluation of Examination Results

The standards for evaluation of examination results are acceptable if they conform to the requirements of ASME Code, Section XI, Subsection IWB-3000, “Acceptance Standards.” The proposed program for repairs of unacceptable indications or replacement of components containing unacceptable indications is acceptable if it is consistent with the requirements of ASME Code, Section XI, Subsection IWA-4000, “Repair/Replacement Activities.” ASME Code, Section XI, Subsection IWB-3000, describes the criteria that establish the need for repair or replacement.

DCD, Tier 2, Section 5.2.4.5, indicates that examination results are evaluated in accordance with ASME Code, Section XI, Subsection IWB-3000, with repairs based on the requirements of Subsection IWA-4000. The staff finds this acceptable because it meets the requirements of ASME Code, Section XI.

5.2.4.3.6 System Leakage and Hydrostatic Pressure Tests

The pressure-retaining ASME Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program meets the requirements of ASME Code, Section XI, Subsection IWB-5000, “System Pressure Tests.”

In DCD, Tier 2, Section 5.2.4.6, the applicant described the system leakage and hydrostatic pressure test requirements. The applicant discusses those portions of ASME Code, Section XI, Subsections IWA-5000, IWB-5000, and IWB-2500, to be applied to system leakage and hydrostatic tests. The staff finds that the specific portions of ASME Code, Section XI, referenced by the applicant are acceptable. However, the staff requested that the applicant revise the DCD to clarify that all applicable requirements of Subsections IWA-5000 and IWB-5000 will apply to system leakage and hydrostatic pressure tests.

In RAI 5.2-65, the staff requested that the applicant revise DCD, Tier 2, Sections 5.2.4.6 and 6.6.6, to clarify that system leakage and hydrostatic pressure tests will meet all requirements of ASME Code, Section XI, Subsections IWA-5000, IWB-5000, IWC-5000, and IWD-5000. The staff tracked RAI 5.2-65 as an open item in the SER with open items. The applicant modified DCD, Section 5.2.4.6 to state that ASME Code Class 1 components will meet the requirements of Subsections IWA-5000 and IWB-5000. Section 6.6 of this report addresses the requirements for ASME Code Class 2 and 3 components. The staff reviewed DCD, Revision 6, and verified that the applicant had made the appropriate modifications to Section 5.2.4.6. RAI 5.2-65 and the associated open item, as they pertain to the RCPB, are resolved.

5.2.4.3.7 Augmented Inservice Inspection To Protect against Postulated Piping Failures

The augmented ISI program for high-energy fluid system piping between containment isolation valves is acceptable if the extent of ISI examinations completed during each inspection interval provides 100-percent volumetric examination of circumferential and longitudinal pipe welds with the boundary of these portions of piping. Section 6.6.3.7 of this report addresses this issue.

5.2.4.3.8 Combined License Information

DCD, Tier 2, Section 5.2.6, includes COL items pertaining to PSI/ISI and the design for accessibility.

In RAI 5.2-64, the staff asked the applicant to revise DCD, Tier 2, Sections 5.2.4 and 6.6, to include a COL applicant item to provide a detailed description of the PSI/ISI programs, augmented inspection programs, and milestones for their implementation. The staff was concerned that the applicant's reference to the COL applicant, did not clearly indicate that the COL applicant must provide, in the COL application, a description of its PSI/ISI program and augmented inspection programs with commitments for their scheduled implementation. The staff understands that the COL licensee will fully develop and implement the actual programs. However, the COL applicant must fully describe the PSI/ISI and augmented inspection programs to allow the staff to make a reasonable assurance finding of acceptability. The staff tracked RAI 5.2-64 as an open item in the SER with open items.

The applicant modified DCD Section 5.2.5 to address the staff's concerns. The staff reviewed DCD, Revision 5, and verified that the applicant had made appropriate modifications to Section 5.2.6. COL applicant Item 5.2-1-A now states that the COL applicant is responsible for providing a full description of the PSI/ISI programs and augmented inspection programs, including milestones for their implementation, by supplementing, as necessary, the information in Section 5.2.4. The requirements described in Sections 5.2.4.1 through 5.2.4.10 are based on ASME Code, Section XI. The staff finds this acceptable because the applicant addressed the staff concerns discussed in RAI 5.2-64 for ASME Code Class 1 systems. Section 6.6 of this report discusses RAI 5.2-64 as it relates to ASME Code Class 2 and 3 systems. RAI 5.2-64 and the associated open item, as it pertains to the RCPB, are resolved.

To address the staff's concerns, expressed in RAI 5.2-62, related to the responsibility of the COL applicant to ensure a design that provides sufficient accessibility to perform PSI/ISI, the applicant modified DCD, Section 5.2.6 to include COL Applicant Item 5.2-3-A. This COL applicant item states that the COL applicant is responsible for developing a plan and providing a full description of its use during construction, PSI, ISI, and design activities for components that are not included in the referenced certified design to preserve accessibility to piping systems to enable NDE of ASME Code Class 1 austenitic and DMWs during ISI (Section 5.2.4). The staff finds this acceptable because the COL applicant will address any design activities beyond the scope of the ESBWR design certification related to ensuring the accessibility of welds for ISI.

5.2.4.4 <u>Conclusions</u>

Based on its evaluation of the system boundary subject to inspection, accessibility, examination categories and methods, inspection intervals, evaluation of examination results, and system leakage and hydrostatic pressure tests, the staff finds that the periodic inspection and testing of the RCPB are acceptable. In addition, the inspection and test program satisfies GDC 32 because it meets the applicable requirements of ASME Code, Section XI, as endorsed in 10

CFR 50.55a. All staff RAIs and open items associated with DCD Section 5.2.4 have been resolved.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

5.2.5.1 Regulatory Criteria

The staff reviewed the RCPB leakage detection system in accordance with SRP Section 5.2.5, Revision 2. Staff acceptance of the leakage detection design is based on its meeting the requirements of the following criteria:

- GDC 2, "Design Basis for Protection against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake

- GDC 30, as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage

5.2.5.2 Summary of Technical Information

In DCD, Tier 2, Revision 7, Section 5.2.5,the applicant described the RCPB leakage detection systems and the design criteria adopted to satisfy NRC regulatory requirements. The systems are designed to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage.

The following are the primary detection methods used for monitoring small unidentified leaks:

- the drywell floor drain high-conductivity waste sump pump activity
- the drywell sump level changes
- the drywell air coolers condensate flow rate
- the fission products' radioactivity

These parameters are continuously monitored and recorded in the MCR and alarmed upon abnormal indications.

The secondary methods used to detect gross unidentified leakage are the pressure and temperature parameters of the drywell atmosphere. High atmospheric pressure in the drywell trips the reactor and initiates isolation of the containment isolation valves. The ambient temperature in the drywell is also monitored and alarmed.

Identified and unidentified leakages from sources within the drywell are collected and directed to separate sumps—the drywell equipment drain low-conductivity waste sump for identified leakages and the drywell floor drain high-conductivity waste sump for unidentified leakages.

In DCD, Revision 7, Tier 2, Section 5.2.6, the applicant identified a COL item as follows.

COL 5.2-2-A, "Leak Detection Monitoring"

The COL Applicant will include in its operating procedure development program:

- Procedures to convert different parameter indications for identified and unidentified leakage into common leak rate equivalents and leak rate rate-of-change values.

- Procedures for monitoring, recording, trending, determining the source(s) of leakage, and evaluating potential corrective action plans.

- Milestone for completing this category of operating procedures.

5.2.5.3 Staff Evaluation

The staff reviewed RCPB leakage detection systems for the ESBWR in accordance with SRP Section 5.2.5, Revision 2. Staff acceptance of the leakage detection design is based on whether the design meets the requirements of GDC 2 and 30. The leakage detection design conforms with GDC 2 if it meets the guidelines of RG 1.29, Revision 4, “Seismic Design Classification,” dated March 2007, Positions C.1 and C.2. The leakage detection design conforms with GDC 30 if it meets the guidelines of RG 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” issued May 1973, Positions C.1 through C.9.

The staff asked the applicant to provide the additional information requested in RAIs 5.2.1 through 5.2.5. The staff reviewed the applicant’s responses and discusses its evaluation below.

In RAI 5.2-1(a), the staff requested the applicant to clarify the statement in DCD, Tier 2, Section 5.2.5, Item (3), stating that the system is equipped with indicators and alarms for each leakage detection system in the control room and permits only “qualitative” interpretations of such indicators. In its response, the applicant stated that the information presented to the MCR operators will be “quantitative,” enabling the operator to convert the various readings to an equivalent leakage rate. The applicant revised the statement in DCD, Tier 2 to reflect that the control room information is both qualitative and quantitative. The NRC staff verified the revised statement in DCD, Tier 2, Section 5.2.5, Revision 3. Therefore, RAI 5.2-1(a) is resolved.

In RAI 5.2-1(b), the staff asked the applicant to explain how the proposed TS and alarm limit for unidentified leakage of 5 gallons per minute (gpm) is consistent with the 1 gpm criterion specified in Positions C.2 and C.5 of RG 1.45. In its response, the applicant maintained its position that the TS and alarm limit for unidentified leakage shall be 5 gpm based on its historical BWR leakage detection and alarm limits. The applicant stated that Positions C.2 and C.5 specified only the “sensitivity” of the instrument rather than the TS or alarm limit and noted that the ESBWR instrument has the sensitivity of 1 gpm. RG 1.45 provides guidance on the “detector sensitivity” and states that “sumps and tanks used to collect unidentified leakage and air cooler condensate shall be instrumented to alarm for increases of from 0.5 to 1.0 gpm.” The staff found that the instrument sensitivity of 1 gpm is neither specified in the TS limit nor reflected by an alarm setpoint that could provide an early warning signal to alert operators to take action. The staff considered the ESBWR alarm limit of 5 gpm alone to be unacceptable because it was inconsistent with RG 1.45, as stated above, and did not serve the intended function of alerting operators to take action before the TS limit is reached. The staff tracked RAI 5.2-1 as an open item in the SER with open items.

In Revision 6 of DCD, Tier 2, Section 5.2.5.5, the applicant stated that a rate-of-change alarm setpoint is established at a lower limit value of 2.2 gpm. The rate-of-change alarm provides an early alert for the control room operators to initiate investigation of the cause and proper response actions for the change of unidentified leakage flow before reaching or exceeding the TS limit. The staff finds that this change addressed the concern identified in RAI 5.2-1. Therefore, RAI 5.2-1 and the associated open item are resolved.

In RAI 5.2-2, the staff asked why ESBWR TS Limiting Condition for Operation 3.4.2 specified a more relaxed limit of 5 gpm for the unidentified RCPB leakage than the limit of 1 gpm specified for the ABWR and for all other advanced reactors. The more relaxed limit could lead to higher operating RCPB leakage rates, fewer RCPB leakage controls, a potentially more humid environment inside containment, and an increased probability of material degradation from corrosion. In response to RAI 5.2-2, the applicant stated that an evaluation of the effects of relative humidity, including that attributable to the proposed leakage limit of 5 gpm, would be part of the equipment qualification requirements in the procurement of equipment. In addition, the applicant stated that the design of the ESBWR has been improved to reduce the likelihood of leaks resulting from SCC, and historically, good operator practice plays a role in the event of an anomaly in unidentified leakage. Typical operator practice will investigate, record, track, and evaluate trends in leakage and take necessary measures to locate, assess, and repair the source of any leakage. The staff agreed that the material design improvement can reduce the likelihood of leaks resulting from SCC, but the improvement cannot eliminate all possible leaks. The staff also agreed that good operator actions at low-level leakage below the TS limit are acceptable measures to address the concern of long-term leakage. To account for the good operator practice, every COL applicant should have operating procedures to manage low-level RCS leakage, and the alarm limit shall be set as low as practicable to provide an early warning signal to the operators to implement the procedures. In conference calls held on August 14, 2006, and January 16, 2007, the applicant agreed to add a COL action item in DCD, Tier 2, Revision 3, Section 5.2.6. This item stated that “operators will be provided with procedures to assist in monitoring, recording, trending, determining the source of leakage, and evaluating potential corrective action.” The staff found the statement unacceptable because it did not indicate that the procedures are for low-level leakage (lower than the TS limit) and did not indicate that the COL holder is responsible for the development of the procedures. In addition, the design needs an appropriate alarm limit (resolution of open item associated with RAI 5.2-1) to provide an early warning signal to the operators to implement the procedures. The staff tracked RAI 5.2-2 as an open item in the SER with open items.

The staff issued supplemental RAI 5.2-2 S03 and RAI 5.2-1 S03 requesting the applicant to address open item associated with RAI 5.2-2. In response to these supplemental RAIs, the applicant revised DCD for the low level leakage alarm set point and COL Information 5.2.6. In Revision 7 of DCD, Tier 2, Section 5.2.5.9, the applicant stated that the licensee is responsible for the development of procedures for monitoring, recording, trending, determining the sources of leakage, and evaluating potential corrective action plans. In DCD, Tier 2, Section 5.2.5.9, the applicant stated that an unidentified leakage rate-of-change alarm provided operators an early alert to initiate response actions before reaching the TS limit. The staff finds that the above changes in Revision 7 of DCD, Tier 2 satisfactorily addressed the concern identified in RAI 5.2-2. Therefore, RAI 5.2-2 and associated open item are resolved.

In RAI 5.2-3, the NRC staff asked the applicant to explain why the TS basis, TS B.3.4.2, “RCS Operational Leakage,” refers to GDC 55, “Reactor Coolant Boundary Penetrating Containment,” but not to GDC 30 as the bases for the TS. GDC 55 discusses the requirements for containment isolation valves, and GDC 30 specifies the quality of the RCPB. In its response, the applicant indicated that it referenced GDC 55 in the context of defining the RCS pressure boundary and referenced GDC 30 in the bases for TS 3.3.4.1, “RCPB Leakage Detection

Instrumentation." The NRC staff reviewed the response and found that, although referencing GDC 55 was acceptable in the context of ESBWR TS B.3.4.2, it was not acceptable without also referencing GDC 30 in TS B.3.4.2. GDC 55 does not require any limit for operational leakage and does not provide any bases for requiring leakage limits, as specified in TS 3.4.2. RG 1.45 provides the guidance for implementing the requirements of GDC 30, and RG 1.45, Position C.9, states that the TS shall include the limiting conditions for identified and unidentified leakage and address the availability of various types of instruments to ensure adequate coverage at all times. This regulatory position, which implements the requirements of GDC 30, provides the bases for both TS 3.3.4.1 and TS 3.4.2. The applicant agreed to revise TS B.3.4.2 by referencing GDC 30 in the bases for TS 3.4.2. The staff confirmed this change in Revision 3 of DCD, Tier 2. Therefore, RAI 5.2-3 is resolved.

In RAI 5.2-4, as it relates to RG 1.45, Position C.7, the staff asked the applicant to clarify whether the procedures that will provide operator guidance on converting leakage instrument indications into a common leakage rate equivalent were generic for the ESBWR design or were to be developed by COL applicants. In its response, the applicant stated that COL applicants would provide the procedures to convert different sources of leakage into a common rate equivalent. This COL action item should be added to DCD, Tier 2, Section 5.2.6. Accordingly, the applicant provided a markup page for Section 5.2.6 in the RAI response. However, when reviewing DCD, Tier 2, Revision 2, the staff could not find the promised COL action item. The applicant agreed to incorporate the change in Revision 3 of DCD, Tier 2. In its review of Revision 3, the staff found that Section 5.2.6 stated that "operators will be provided with a procedure to determine the identified and unidentified leakage in order to establish whether the leakage rates are within the allowable TS." The staff found this statement unacceptable for two reasons. First, the statement should identify the COL holder as responsible for the development of the procedures, and second, the statement shall better characterize the purpose of the procedures. The purpose is to convert different sources of leakage (such as sump pump activity, sump level, condensate flow rate, and radioactivity) into a common rate equivalent (expressed in gpm). Operators can use this leak rate information to monitor the leakage and to keep the leakage well below the TS limit. The purpose of the procedures is not limited to establishing whether the leakage rates are within the allowable TS. RAI 5.2-4 was being tracked as confirmatory item in the SER with open items.

In Revision 7 of DCD, Tier 2, Section 5.2.5.9, the applicant stated that the licensee is responsible for the development of a procedure to convert different parameter indications for identified and unidentified leakage into common leak rate equivalents and leak rate rate-of-change values. In DCD, Tier 2, Section 5.2.5.9, the applicant stated that typical monitoring includes parameters such as sump pump run time, sump level, condensate transfer rate, and process chemistry/radioactivity. Furthermore, the monitored leakage equivalents provide information used by the plant operators to manage the leakage, establish whether the leakage rates are within the allowable TS, and determine the trend. The staff finds that the changes in Revision 7 of DCD, Tier 2 addressed the staff concern identified above. Therefore, RAI 5.2-4 and associated confirmatory item are resolved.

In RAI 5.2-5, as it relates to the capability of the leak detection instrument to maintain and perform its safety functions following an earthquake, the staff asked the applicant to clarify which of the leak detection instrumentation discussed in DCD, Tier 2, Section 5.2.5.2, is required to perform the containment isolation function and which is not. The leak detection instrumentation required to perform the isolation function is classified as Class 1E, seismic

Category I, and therefore, should be consistent with the guidelines of RG 1.29. In its response, the applicant revised DCD, Tier 2, Section 5.2.5, to identify the leak detection instruments that are used for isolation functions and the instruments that are not used for isolation functions. The staff confirmed that this modification appeared in DCD, Tier 2, Revision 6. In addition, the applicant stated that, with one exception, leak detection instruments that are not required for isolation functions are not required to remain functional following an earthquake. The exception is the drywell fission product radiation monitoring system, which is seismically qualified and should be designed in a manner that is consistent with the guidance of Positions C.1 and C.2 of RG 1.29. The staff found the applicant's response acceptable and determined that the ESBWR design satisfies Position C.6 of RG 1.45 and Positions C.1 and C.2 of RG 1.29. Therefore, the design satisfies GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake. RAI 5.2-5 is resolved.

The staff reviewed DCD, Tier 2, Chapter 16, relating to the TS of the RCPB leakage detection and issued RAI 16.2-1 (gaseous radiation monitor) and RAI 16.2-4 (rate-of-change limit in RCS operational leakage). Chapter 16 of this report discusses the responses to, and resolution of, the aforementioned RAIs.

5.2.5.4 Conclusions

Based on the above, the staff finds that the applicant has met the requirements of GDC 2 with respect to the systems' capability to maintain and perform their safety functions in the event of an earthquake by meeting Positions C.1 and C.2 of RG 1.29 and the requirements of GDC 30, as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The staff reviewed DCD, Tier 2, Section 5.3.1, Revision 7, in accordance with SRP Section 5.3.1, Revision 2. The applicant's RV materials are acceptable if they meet codes, standards, and regulatory guidance commensurate with the safety function to be performed. This will ensure that the materials meet the relevant requirements of 10 CFR 50.55a; Appendix G and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50; and GDC 1, 4, 14, 30, 31, and 32. These requirements are discussed below:

- GDC 1 and 30 and 10 CFR 50.55a(a)(1) require SSCs important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 4 requires SSCs important to safety to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

- GDC 14 requires the RCPB to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 31 requires the RCPB to be designed with sufficient margins to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and minimize the probability of rapidly propagating fracture.

- GDC 32 requires the RCPB components to be designed to permit an appropriate material surveillance program for the RV.

- Appendix G to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviewed the RV materials as they relate to the materials testing and acceptance criteria for fracture toughness contained in Appendix G. In accordance with Appendix G, the RV beltline materials must have Charpy upper-shelf energy (USE) values, in the transverse direction for base material and along the weld for weld material, of no less than 101.7 N-m (75 foot-pound (ft-lb)) initially and must maintain Charpy USE values throughout the life of the vessel of no less than 67.8 N-m (50 ft-lb).

- Appendix H to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in the fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance to ASTM E-185, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels." Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine whether they meet the relevant requirements of Appendix H for determining and monitoring material fracture toughness.

5.3.1.1 Summary of Technical Information

5.3.1.1.1 Material Specifications

The applicant indicated that the material specifications are consistent with ASME Code requirements. All ferritic RV materials must comply with the fracture toughness requirements of 10 CFR 50.55a and Appendices G and H to 10 CFR Part 50.

DCD, Tier 2, Table 5.2-4, identifies the materials used in the RV and appurtenances. The chemical compositions of the ferritic materials of the RV beltline are restricted to the maximum limits shown in DCD, Tier 2, Table 5.3-1. Copper, nickel, and phosphorus content is restricted to reduce sensitivity to irradiation embrittlement in service.

5.3.1.1.2 Special Processes Used for Manufacturing and Fabrication

The RV is constructed primarily from low-alloy, high-strength steel plate and forgings. Plates are ordered to ASME Code SA-533, Type B, Class 1, and forgings to ASME Code SA-508, Grade 3, Class 1, specifications. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low-alloy steels. The shells and vessel heads are made from formed plates or forgings, whereas flanges and nozzles are made from forgings. Welding performed to join these vessel components is consistent with procedures qualified in accordance with the requirements of Sections III and IX of the ASME Code. GTAW, GMAW, shielded metal arc welding, and submerged arc welding (SAW) processes may be employed. Electroslag welding is not used except for cladding.

Postweld heat treatment of all low-alloy welds is performed in accordance with ASME Code, Subsection NB-4620 (DCD, Tier 2, Table 5.3-1). The materials, fabrication procedures, and testing methods used in the construction of the ESBWR RV meet or exceed the requirements of ASME Code, Section III, Class 1 vessels.

The RV assembly components are classified as ASME Code Class 1. Complete stress reports on these components are prepared in accordance with ASME Code requirements. NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," issued November 1980, is also considered for FW nozzle and other such RV inlet nozzle designs. Action Plan Item A-10, "BWR Feedwater Nozzle Cracking," is considered resolved through compliance with NUREG-0619, consistent with the NRC resolution, and compliance with GL 81-11, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking (NUREG-0619)."

The staff's review of DCD, Tier 2, Section 5.3.1, in accordance with SRP Section 5.3.1, identified areas in which additional information was necessary to complete the review of the RV materials.

In response to RAI 5.3-1, the applicant confirmed that the carbon content is limited so as not to 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 (200 degrees F). The applicant also stated that in Table 5.2-4 strength is the only distinction between 304 and 304L/316 and 316L. The applicant updated DCD Table 5.2-4 indicating that for these components the maximum allowable carbon content is 0.02 percent. The applicant's response and revision to DCD, Table 5.2-4 were acceptable to the staff. RAI 5.3-1 is closed.

In response to RAI 5.3-2, the applicant confirmed that the current practice for welding stub tubes to the bottom head is automatic GTAW. The inclusion of manual welding in the DCD is to allow for local repair using manual GTAW or GMAW. The applicant also confirmed that all weld metal is Alloy 82 with stabilization parameter control. Use of Alloy 182 is prohibited in components that come into contact with reactor water. The applicant modified the appropriate DCD sections accordingly. The applicant's response and revisions to the DCD were acceptable to the staff. RAI 5.3-2 is closed.

In response to RAI 5.3-3, the applicant stated the following:

- Several BWR RVs have been site assembled. This includes Vermont Yankee, Monticello, Leibstadt, Clinton, and Limerick. The process for ESBWR has not been finalized at this time, but it is anticipated that the nearly completed RV will be shipped to the site in two or possibly three sections. Joining of the sections at the site may be done with the vessel axis vertical using mechanized welding equipment. Alternately, temporary rollers may be set up at the site, and the closure weld completed with mechanized SAW or GMAW.

- Local post-weld heat treatment, as allowed by ASME Code, Section III, will be performed on the circumferential weld(s). This is a relatively

> simple operation because the weld joins two axisymmetric cylinders of uniform thickness. The goal is to locate the welds away from discontinuities. Finite-element analysis will be used to establish the heating pattern and define temperature gradients away from the heated band. This will be followed by stress analysis to demonstrate that stresses in the adjacent material are maintained at acceptable levels. This approach has previously been successfully used to apply local post-weld heat treatment to RV nozzles where reapplication of nozzle butters was required. Likewise, this approach is routinely used to attach main steam nozzle extension forgings of low alloy steel to the steam nozzle at the ABWR construction sites. Local heat treatment of the final closing weld has been standard practice by some European manufacturers (e.g., the Cofrentes RPV) since most of their furnaces do not have the capacity to heat treat a complete RPV. The local heat treatments were performed using either heating pads or induction heating.

In its response to RAI 5.3-3, the applicant also confirmed that the process of assembling the RV at a plant site has not yet been finalized. Thus, the staff finds that the fabrication process and examination process will be verified using the ITAAC described in DCD, Tier 1, Table 2.1.1-2, Item 5.

5.3.1.1.3 Special Methods for Nondestructive Examination

The NDE of the RV and its appurtenances is conducted in accordance with the requirements of ASME Code, Section III. Volumetric examination and surface examination are performed on all pressure-retaining welds, as required by ASME Code, Section III, Subsection NB-5320. In addition, all pressure-retaining welds are given a supplemental ultrasonic preservice examination in accordance with ASME Code, Section XI. The ultrasonic examination method, including calibration, instrumentation, scanning, and coverage, is based on the requirements of ASME Code, Section XI, Appendix I.

5.3.1.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

DCD, Tier 2, Section 5.3.1.4, addresses issues raised in the following RGs affecting the RV:

- RG 1.31 addresses controls on stainless steel welding, which are discussed in DCD, Tier 2, Section 5.2.3.4.2.

- RG 1.34 is not applicable to the ESBWR vessel because electroslag welding is not used in structural low-alloy steel welds.

- RG 1.43 is not applicable to the ESBWR vessel because the RV is constructed from low-alloy steel forgings or plates conforming to the SA-508, Grade 3, or SA-533, Type B, specification which are produced to fine grain practice. Therefore, underclad cracking is not a concern.

- RG 1.44 addresses the control of sensitization of stainless steel by the use of service-proven low-carbon materials and appropriate design and processing steps, including solution heat treatment, control of welding heat input, control of heat treatment during fabrication, and control of stresses.

- RG 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Code, Sections III and IX. The preheat temperature employed for welding of low-alloy steel meets or exceeds the recommendations of ASME Code, Section III, Appendix D. Components are either held for an extended time at preheat temperature to ensure removal of hydrogen or preheat is maintained until postweld heat treatment.

- RG 1.71 addresses welder qualification for areas of limited accessibility, which is addressed in DCD, Tier 2, Section 5.2.3.4.2.

- RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," issued May 1988, addresses ways to predict changes in transition temperature and USE.

- RG 1.37 addresses the quality assurance requirements for the cleaning of systems and components on the site during and at the completion of construction. This cleaning follows written procedures which provide for cleanliness and ensure that the components are not exposed to materials or practices that may degrade their performance. For components containing stainless steel, RG 1.37 presents the procedures. The procedures prohibit contact with low-melting-point compounds and substances that are known to cause SCC or that can release, in any manner, substances that can cause such problems. In addition, controls are placed on the use of grinding wheels and wire brushes, which ensures that they cannot introduce degrading materials either through prior usage or through their materials of construction. In this context, degradation includes SCC. Controls also limit the introduction of unnecessary dirt and require restrictions on dirt-producing processes, such as welding or grinding, which include prompt cleaning.

The staff found that the applicant's use of the RGs, mentioned above, to ensure the integrity of the RV is acceptable.

5.3.1.1.5 Fracture Toughness

In DCD Section 5.3.1.5, the applicant described the methods, Codes, and Standards used to comply with the requirements for fracture toughness testing in Appendix G to 10 CFR Part 50. Specifically the applicant addressed the material test coupons, location and orientation of test specimen, records and procedures for impact testing, Charpy curves for the RPV beltline, bolting material, and fracture toughness margins to control reactivity. Staff's evaluation is provided in Section 5.3.1.2 of this report.

5.3.1.1.6 Material Surveillance

Appendix H to 10 CFR Part 50 presents the requirements for a material surveillance program for operating reactors. The purpose of the material surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the RV beltline region, which result from exposure of these materials to neutron irradiation. Material surveillance is accomplished using surveillance capsules, which are holders of archival beltline material and fast neutron (i.e., neutrons with energy greater than 1.0 million electron volts (MeV) dosimeters. Assessment of the irradiated material samples yields a measure of the embrittlement, and measurement of the dosimeter activation estimates the irradiation exposure.

RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” issued March 2001, which is based on GDC 14, 30, and 31, describes methods and practices acceptable to the staff regarding calculational techniques and statistical practices using the dosimetry measurements. In addition, the results of the dosimetry are used to benchmark and validate calculational methods for estimating vessel irradiation.

DCD, Section 5.3.1.6.1 states that RV material surveillance specimens are provided in accordance with the requirements of ASTM E-185 and Appendix H to 10 CFR Part 50. Materials for the program are selected to represent materials used in the reactor beltline region. Specimens are manufactured from a forging actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld HAZ material. The base metal and weld are heat treated in a manner that simulates the actual heat treatment performed on the beltline region of the completed vessel. Each in-reactor surveillance capsule contains 36 Charpy V-notch and 6 tensile specimens. The capsule loading consists of 12 Charpy V-notch specimens each of base metal, weld metal, and HAZ material and 3 tensile specimens each from base metal and weld metal. A set of out-of-reactor beltline Charpy V-notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. Neutron dosimeters and temperature monitors are located within the capsules, as required by ASTM E-185.

Four capsules are provided to monitor the 60-year design life of the vessel. This exceeds the three capsules specified in ASTM E-185, as required by Appendix H to 10 CFR Part 50, since the predicted transition temperature shift is less than 55.6 degrees C (100 degrees F) at the inside surface of the low-alloy steel vessel.

The following proposed withdrawal schedule is modified from the ASTM E-185 schedule to monitor the RV for its 60-year design life:

- First capsule: After 6 effective full-power years (EFPY)
- Second capsule: After 20 EFPYs
- Third capsule: With an exposure not to exceed the peak end of life (EOL) fluence
- Fourth capsule: Schedule to be determined based on results of first three capsules, in accordance with ASTM E-185, paragraph 7.6.2

In response to RAI 5.3-4, the applicant explained that achieving a lead factor exceeding 1.0 is relatively easy in the ESBWR because there are no obstructions in the annulus that restrict placement of the capsule holders. The location of the axial and circumferential flux peaks are known from fluence calculations, and the capsule holders can be placed precisely at these peak locations (there are a total of eight peak locations). Since the capsule holder is mounted somewhat inboard of the vessel wall, a lead factor greater than 1.0 is assured. The applicant modified DCD, Tier 2, Section 5.3.4, Revision 3 and confirmed that the COL applicant will identify the following information:

- specific materials in each surveillance capsule
- capsule lead factors
- withdrawal schedule for each surveillance capsule

- neutron fluence to be received by each capsule at the time of its withdrawal
- vessel EOL peak neutron fluence

In response to RAI 5.3-5, the applicant stated that, like all BWRs, the ESBWR will operate at a nominal temperature of about 228 degrees C (550 degrees F). The temperature of the RV wall inside surface may be slightly, but not significantly, lower and certainly not below 525 degrees F.

The applicant also modified DCD, Tier 2, Section 5.3.1.6.1, of the DCD and included a statement that the RV material will not be exposed to normal operating temperatures below 274 degrees C (525 degrees F) and that enhanced radiation embrittlement resulting from temperature is of no concern.

The applicant's schedule for removing the capsules for postirradiation testing includes the withdrawal of four capsules, in accordance with ASTM E-185-82 (i.e., the 1982 edition of ASTM E-185) and Appendix H to 10 CFR Part 50. Staff's evaluation is provided in Section 5.3.1.2 of this report.

5.3.1.1.7 Reactor Vessel Fasteners

As described in DCD Table 5.3-1, the materials for the fasteners for the RV are controlled as follows:

- Closure studs, nuts, and washers for the main closure flange are composed of ASME Code SA-540, Grade B23 or Grade B24 material with a minimum yield strength level of 893 MPa (129.5 kilo pound/square inch (ksi)).

- Maximum measured ultimate tensile strength of the stud bolting materials must not exceed 1172 MPa (170 ksi).

5.3.1.2 Staff Evaluation

The staff reviewed DCD, Tier 2, Section 5.3.1, Revision 7, in accordance with SRP Section 5.3.1.

The NRC staff also reviewed the ESBWR RV materials to ensure that they meet the relevant requirements of GDC 1 and 30 and 10 CFR 50.55a(a)(1), as they relate to material specifications, fabrication, and NDE. Compliance with these requirements will determine whether the RV materials are adequate to ensure a quality product commensurate with the importance of the safety function to be performed. The material specifications for the ESBWR design are consistent with the requirements of ASME Code, Section III, and Appendix G to 10 CFR Part 50. In addition, the design and fabrication of the RV conforms to the requirements of ASME Code, Section III, Class 1. Furthermore, the RV and its appurtenances are fabricated and installed in accordance with ASME Code, Section III, Subsection NB-4100. The NDE of the RV and its appurtenances is conducted in accordance with ASME Code, Section III, requirements. Examination of the RV and its appurtenances by NDE complies with ASME Code, Section III, Subsection NB-5000. The applicant stated that all plates, forgings, and bolting are 100-percent ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods, as required by ASME Code, Division 1, Section III.

The staff finds this acceptable because compliance with the requirements of ASME Code, Section III, and Appendix G to 10 CFR Part 50 constitutes an adequate basis for satisfying the requirements of GDC 1 and 30 and 10 CFR 50.55a(a)(1) as they relate to the material specifications, fabrication, and NDE of RV materials.

Section 5.2.3 of this report provides the staff's evaluation of the welding of ferritic steels and austenitic stainless steels and addresses GDC 4.

DCD, Tier 2, Table 5.3-1, provides the maximum limits for the elements in the materials of the RV beltline. Specified limits for RV materials used in the core beltline region are the following:

- Base Materials: 0.05-percent maximum copper, 0.006-percent maximum phosphorus, 1.0-percent maximum nickel (forging), and 0.73-percent maximum nickel (plate)

- Weld Materials: 0.05-percent maximum copper, 0.008-percent maximum phosphorus, 1.0-percent nickel, and 0.008-percent maximum vanadium

Table 5.3-1 also provides the maximum limits for the RV studs, nuts, and washers for the main closure flange.

The tests for fracture toughness of RV materials specified in the DCD are consistent with ASME Code, Section III, Subsection NB-2300, and Appendix G to 10 CFR Part 50. The staff confirmed that the applicant's initial Charpy V-notch minimum upper-shelf fracture energy levels for the RV beltline base metal transverse direction and welds are 101.7 N-m (75 ft-lb). DCD, Tier 2, Table 5.3-3, indicates that the EOL values for the USE are greater than 67.8 N-m (50 ft-lb) for the beltline forgings and welds. The staff confirmed this by using the calculations of RG 1.99 for the beltline forgings and welds. The predicted EOL Charpy USE and adjusted reference temperature (ART or adjusted reference temperature nil ductility (RT_{NDT})) for the RV materials comply with the requirements of Appendix G to 10 CFR Part 50. The fracture toughness tests required by the ASME Code and Appendix G provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RV. This methodology will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with the provisions of Appendix G to 10 CFR Part 50 satisfies the requirements of GDC 14 and 31 and 10 CFR 50.55a regarding the prevention of fracture of the RV. Therefore, the staff finds that the applicant has adequately met the requirements of GDC 14 and 31 and 10 CFR 50.55a for the RV.

The design of a RV must consider the potential embrittlement of RV materials as a consequence of neutron irradiation and the thermal environment. GDC 32 requires that the RCPB components be designed to permit an appropriate material surveillance program for the RV. Appendix H to 10 CFR Part 50 details the requirements of such a program.

The applicant explained that, since there are no obstructions in the annulus that restrict placement of the capsule holders, achieving a lead factor of greater than 1.0 is relatively easy in the ESBWR. The location of the axial and circumferential flux peaks are known from fluence calculations, and the capsule holders can be placed precisely at these peak locations (there is a total of eight peak locations). Mounting the capsule holder somewhat inboard of the vessel wall ensures a lead factor greater than 1.0. The applicant also confirmed that it will perform an analysis defining the lead factors and the azimuth locations of the surveillance holders. DCD, Tier 2, Section 5.3.4 includes this as a COL action item.

To meet the requirements of GDC 32, the ESBWR design includes provisions for a material surveillance program to monitor changes in the fracture toughness caused by exposure of the RV beltline materials to neutron radiation. Appendix H to 10 CFR Part 50 requires that the surveillance program for the ESBWR RV meets the recommendations of ASTM E-185. ASTM E-185 applies to plants designed for a 40-year life, whereas the design life of the ESBWR is 60 years. ASTM E-185 recommends a minimum of three surveillance capsules for an RV with an EOL shift of less than 38 degrees C (100 degrees F). The ESBWR surveillance capsule program includes four specimen capsules, with archive materials available for additional replacement capsules. The staff verified that the surveillance test materials will be prepared from samples taken from the materials used in fabricating the beltline of the RV. In addition, the staff verified that the base metal, weld metal, and HAZ materials included in the program will be those predicted to be most limiting in terms of setting pressure-temperature (P/T) limits for operation of the reactor to compensate for radiation effects during its lifetime. The staff finds that the materials selection, withdrawal, and testing requirements for the ESBWR design are consistent with those recommended in ASTM E-185-82. Compliance with the materials surveillance requirements of Appendix H to 10 CFR Part 50 and ASTM E-185 satisfies the requirements of GDC 32 for an appropriate surveillance program for the RV. Thus, the ESBWR design meets the requirements of GDC 32.

The applicant indicated that the material used to fabricate the closure studs will meet the fracture toughness requirements of Section III of the ASME Code and Appendix G to 10 CFR Part 50. NDE of the studs will be performed according to Section III of the ASME Code, Subsection NB-2580. In addition, ISI will be performed according to Section XI of the ASME Code, supplemented by Subsection NB-2545 or NB-2546. Conformance with the recommendations of RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs,” issued October 1973, ensures the integrity of the ESBWR RV closure studs and satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a. Compliance with the recommendations of RG 1.65 also satisfies the requirement in GDC 31 for the prevention of fracture of the RCPB and the requirements of Appendix G to 10 CFR Part 50, as detailed in the provisions of Section III of the ASME Code.

Generic Letter GL 92-01

GL 92-01, “Reactor Vessel Structural Integrity,” addressed NRC concerns regarding compliance with the requirements of Appendices G and H to 10 CFR Part 50, which address fracture toughness requirements and RVSP requirements, respectively. Specifically, NRC had concerns about Charpy upper shelf energy predictions for end of life for the limiting beltline weld and the plate or forging, RVs constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition, and use of RG 1.99, Revision 2, to estimate the embrittlement of the materials in the RV beltline. In addition, the NRC was concerned about RVSP compliance with ASTM E

185, which requires that the licensee take sample specimens from actual material used in fabricating the beltline of the RV.

The ESBWR DCD, Revision 5, states that the RV materials surveillance specimens are provided in accordance with the requirements of ASTM E-185. Section 5.3.1.6.1 states that the materials for the program are selected to represent materials used in the reactor beltline region and that the specimens are manufactured from forgings actually used in the beltline region and a weld typical of those in the beltline region, thus representing base metal, weld material and the weld HAZ material. Therefore, the applicant has addressed the entire beltline region in their RVSP. The DCD also states that the predictions for changes in transition temperature and upper shelf energy are made in accordance with the requirements of RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." In addition,

Finally, COL applicants referencing the ESBWR DCD are required to develop a description of their RVSP that will include (1) specific materials in each surveillance capsule; (2) capsule lead factors; (3) withdrawal schedule for each surveillance capsule; (4) neutron fluence to be received by each capsule at the time of its withdrawal; and, (5) vessel end-of- life peak neutron fluence (COL item 5.3.2-A).

The staff finds that the applicant has met the intent of GL 92-01. In addition, a COL applicant that incorporates by reference the ESBWR DCD and provides an acceptable response to the COL items should also meet the intent of the GL. Furthermore, a COL applicant will continue to meet the intent of the GL in the future by providing the summary test reports, in accordance to ASTM E 185-82, to the NRC upon withdrawal of each surveillance capsule.

Task Action Plan Item A-10

As discussed in NUREG-0933, Task Action Plan Item A-10 addresses the issue of cracks found during the inspection of the FW nozzles of 20 RVs. It was determined that cracking was due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low FW temperature when flow is unsteady and intermittent. Once initiated, the cracks enlarge from high pressure and thermal cycling associated with startups and shutdowns.

ESBWR DCD, Revision 5, Section 3.9.3.2 states that RPV assembly components are classified as ASME Class 1. Complete stress reports on these components are prepared in accordance with the Code requirements. The guidance from NUREG-0619 and associated GLs 80-95 and 81-11 is factored into the FW nozzle and sparger design. The FW nozzle/sparger design does not allow incoming FW flow to have direct contact with the nozzle bore region, and the double thermal sleeve design adds further protection against thermal cycling on the nozzle. The action plan item A-10, "BWR feedwater Nozzle Cracking," is considered resolved through compliance with NUREG-0619 and is consistent with the NRC resolution. The resolution of this Task Action Item is also documented in Table 1.11-1 of the DCD. Therefore, the staff finds the applicant has appropriately addressed this issue.

Task Action Plan Item A-11

As discussed in NUREG-0933, Task Action Plan Item A-11 addresses the issue that, because of the remote possibility that nuclear RPVs designed to the ASME Code might fail, the design of nuclear facilities must provide protection against RV failure.

Prevention of RV failure depends primarily on maintaining the RV material fracture toughness at levels that will resist brittle fracture during plant operation. As plants accumulate more service time, neutron irradiation reduces the material fracture toughness and initial safety margins. This issue was resolved with the publication of NUREG-0744, "Resolution of the Task A-11, Reactor Vessel Materials Toughness Safety Issue," Revision 1, dated October 1982, GL 82-26, "NUREG-0744, Revision 1, Pressure Vessel Material Fracture Toughness," dated November 12, 1981. This issue did not result in establishing new regulatory requirements.

DCD, Section 5.3.1.1 states that the ESBWR RV design complies with the provisions of ASME Section III, and should also meet the requirements of ASME Code Section II materials and Appendix G to10 CFR Part 50. The fracture toughness tests required by these regulations provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. In addition, Table 1.11-1 addresses Task Action Plan Item A-11 by stating that the issue is resolved through compliance with NUREG-0744. Therefore, the staff finds the applicant has appropriately addressed this issue.

Issue 111: Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments

As discussed in NUREG-0933, Issue 111 addresses stress corrosion cracking of ferritic steels. The cracks, first detected on the shell side of steam generator #32 of Indian Point Unit 3 (IP-3), were found to be caused by a low cycle corrosion fatigue phenomenon with cracks initiating at areas of localized corrosion and propagating by fatigue. The cause of the pitting/cracking was considered to be related to high dissolved oxygen levels and copper species in solution. Further testing indicated that the water chemistry control at IP-3 has been poor for some time.

Investigation of recent history current BWR vessels and the proposed design of ESBWR reveal that no stress corrosion cracking was ever observed in low alloy steel. The ESBWR vessel is claded with stainless steel or Ni-Cr-Fe alloy and will go through ASME Section XI inspection (Section 5.3.3.3). Also, there will be no copper tubing in the ESBWR heat exchangers and therefore there will be no copper species in the reactor water solution as was found in the IP-3 steam generator. Finally, the ESBWR reactor water cleaning/shutdown cooling system will measure conductivity, dissolved oxygen, pH, chloride, silica, etc. as part of a the sampling program requirement described in SRP Section 9.3.2 (ESBWR DCD, Tier 2, Table 9.3-1). Therefore, the staff finds that Issue 111 is not applicable to the ESBWR vessel.

5.3.1.3 Conclusions

The staff finds that the ESBWR RV material specifications, RV manufacturing and fabrication processes, NDE methods of the RV and its appurtenances, fracture toughness testing, material surveillance, and RV fasteners are acceptable and meet the material testing and monitoring requirements of Section III of the ASME Code; Appendices G and H to 10 CFR Part 50; and 10 CFR 50.55a, which provide an acceptable basis for satisfying the requirements of GDC 1, 14, 30, 31, and 32.

5.3.2 Pressure-Temperature Limits

The staff reviewed DCD, Tier 2, Section 5.3.2, Revision 7, in accordance with SRP Section 5.3.2, Revision 2. The applicant's P/T limit curves are acceptable if they meet codes, standards, and regulatory guidance commensurate with the safety function to be performed. This will ensure that the applicant meets the relevant requirements of 10 CFR 50.55a; Appendix G to 10 CFR Part 50; and GDC 1, 14, 31, and 32. These requirements are discussed below:

- GDC 1 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

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

- GDC 31 requires that the RCPB be designed with sufficient margin to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and minimize the probability of rapidly propagating fracture.

- GDC 32 requires that the RCPB components be designed to permit an appropriate material surveillance program for the RV.

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the RCPB in nuclear power plants. The staff evaluates the P/T limit curves based on Appendix G to 10 CFR Part 50, RG 1.99, and SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 requires that P/T limit curves for the RV be at least as conservative as those obtained by applying the methodology of ASME Code, Section XI, Appendix G.

RG 1.99 contains methodologies for determining the increase in transition temperature and the decrease in USE resulting from neutron radiation. SRP Section 5.3.2 provides an acceptable method of determining the P/T limit curves for ferritic materials in the beltline of the RV based on the linear elastic fracture mechanics methodology of ASME Code, Section XI, Appendix G. The basic parameter of this methodology is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration ASME Code, Section XI, Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions; for hydrostatic testing curves, Appendix G to the ASME Code requires a safety factor of 1.5.

The methods of Appendix G to the ASME Code postulate the existence of a sharp surface flaw in the RV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-quarter of the RV beltline thickness and a length equal to 1.5 times the RV beltline thickness. The critical locations in the RV beltline region for calculating heatup and cooldown P/T curves are the one-quarter thickness (1/4T) and three-quarter thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw if initiated and grown from the inside and outside surfaces of the RV, respectively.

The ASME Code, Section XI, Appendix G, methodology requires that applicants determine the limiting materials' ART. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent on the amount of copper and nickel in the material and may be determined either from tables in RG 1.99 or from surveillance data. The fluence factor depends on the neutron fluence at the maximum postulated flaw depth. The margin term depends on whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99 or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculational procedures. RG 1.99 describes the methodology for calculating the margin term.

Appendix H to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with ASTM E-185-82. Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine whether they meet the relevant requirements of Appendix H as they relate to determining and monitoring material fracture toughness.

In addition, RG 1.190 describes attributes of vessel fluence calculational methodologies (or equivalent) that are acceptable to the staff.

5.3.2.1 Summary of Technical Information

DCD, Tier 2, Section 5.3, describes material properties and the effects of irradiation on material fracture toughness and the irradiation surveillance requirements. DCD, Tier 2, Section 5.3.1.6.5, references DCD, Tier 2, Section 4.1.4.5. Reference is made to neutron dosimetry measurements for the purpose of establishing a relationship between activation and reactor thermal power. DCD, Tier 2, Section 5.3.3, outlines the vessel design bases for material construction, fabrication, inspection, operating conditions, inservice surveillance, safety design, and power generation. The section continues with a description of RV internals, CRD housing, in-core neutron flux monitoring, RV insulation, and RV nozzle design and inspections. The section concludes with fabrication methods, inspection requirements, and the 10 CFR 50.55a vessel requirements.

The applicant's responses to RAIs 5.3-9 to 5.3-12 and 5.3-14 supplement this information. The additional information describes vessel dosimetry and compliance with Appendix H to 10 CFR Part 50 and ASTM-E185; eliminates the reference to RG 1.2, which the NRC has withdrawn; restates Section 5.3.3; corrects a typographical error on vessel failure probability; and expands Table 5.3-4 to add peak vessel fluence and its azimuthal location. Finally, in its response to RAI 5.3-14, the applicant committed to add text to DCD, Revision 4, Section 4.1.4.5, to clarify that the application of the staff-approved vessel fluence methodology (NEDC-32983P-A, "GEH Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," issued August 2000) will include resonance self-shielding in the preparation of the working cross-section library.

The ESBWR DCD discussion on P/T limits indicates that the heatup and cooldown P/T limit curves are required as a means of protecting the RV during startup and shutdown to minimize the possibility of brittle fracture. The P/T limits are based on flaw sizes specified in Subsection G-2120 of ASME Code, Section III, Appendix G, "Protection Against Nonductile Failure." The maximum throughwall temperature gradient from continuous heating or cooling at 55.6 degrees C (100 degrees F) per hour was considered. The safety factors applied were those specified in ASME Code, Section III, Appendix G. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the ART, which includes a reference nil ductility temperature shift, initial RT_{NDT}, and margin. The initial RT_{NDT} of the vessel materials is determined in accordance with the methodology presented in ASME Code, Section III, Subsection NB-2320; DCD, Tier 2, Table 5.3.1, lists the requirements.

The applicant evaluated the RV flange, RV head and flange areas, FW nozzles, bottom head, and core beltline areas; the operating limit curves are based on the most limiting locations. The P/T limits are based on flaw sizes specified in Subsection G-2120 of ASME Code, Section III, Appendix G. The analysis considered the maximum throughwall temperature gradient from continuous heating or cooling at 55.6 degrees C (100 degrees F) per hour. The safety factors applied were those specified in ASME Code, Section III, Appendix G.

The applicant stated that the P/T curves are developed considering a radiation embrittlement of up to 54 EFPYs. This is consistent with an expected plant life of 60 years, with a 90-percent load factor. The maximum chemical compositions for the RV materials used in the core beltline region are as follows:

- Base materials: 0.05-percent maximum copper, 0.006-percent maximum phosphorus, 1-percent maximum nickel (for forging), and 0.73-percent nickel (for plate)

- Weld materials: 0.05-percent maximum copper, 0.008-percent maximum phosphorus, 1-percent maximum nickel, and 0.05-percent maximum vanadium

The following shows the results of the RV fluence analysis for the EOL:

- Expected peak EOL fluence at the 1/4T location: Less than 1.37×10^{19} neutrons per square centimeter (n/cm^2) (E greater than 1.0 MeV)

- Expected 1/4T EOL fluence for the weld above the top of the active fuel: Less than 4.14×10^{17} n/cm^2 (E greater than 1.0 MeV)

The operating curves are developed in accordance with Appendix G to 10 CFR Part 50. The initial RT_{NDT} for all RV materials is -20 degrees C (-4 degrees F). Thus, a minimum flange and boltup temperature of RT_{NDT} plus 33 degrees C (60 degrees F) or 13 degrees C (56 degrees F) will be used for tensioning at preload condition and during detensioning. In DCD, Tier 2, Figures 5.3-1 and 5.3-2, the applicant provided generic curves for the ESBWR RV design. These are limiting curves based on the maximum copper and nickel contents and EOL peak fluence.

The results of the material surveillance program will verify the validity of ΔRT_{NDT} used in the calculation for the development of heatup and cooldown curves. The projected fluence, copper content, and nickel content, along with the RT_{NDT} calculation, will be occasionally adjusted, if necessary, using the surveillance capsules.

The applicant also indicated that temperature limits for core operation (both critical and noncritical), inservice leak tests, and hydrotests are calculated in accordance with ASME Code, Section XI, Appendix G.

5.3.2.2 Staff Evaluation

DCD, Tier 2, Section 5.3.1.6.3, references DCD, Tier 2, Section 4.1.4.5, which also does not identify the proposed calculational method, but in turn, references DCD, Tier 2, Section 12.3. The material in Section 12.3 is also irrelevant to the subject. The material reviewed below is from the applicant's responses to the staff's RAIs.

DCD, Tier 2, Section 5.3.5, lists the applicant-approved fluence methodology topical report NEDC-32983P-A. The staff's evaluation established that this methodology is consistent with the guidance in RG 1.190 and, therefore, is acceptable.

The response to RAI 5.3-9 regarding DCD, Tier 2, Section 5.3.1.6.5, clarifies that the dosimetry will satisfy the requirements of Appendix H to 10 CFR Part 50 and ASTM-185, adopted by reference in Appendix H. The proposed set of dosimeters provides adequate spectral coverage. It is noteworthy that jet pumps or other obstructions will not shadow the dosimetry in the ESBWR (as is the case with current BWRs).
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In its response to RAI 5.3-10, the applicant proposed a revised DCD, Tier 2, Section 5.3.3, which, among other things, eliminated reference to RG 1.2, which the NRC has withdrawn; corrected a typographical error regarding vessel failure probability (as noted in RAI 5.3-11); and restated DCD, Tier 2, Section 5.3.3.1. Finally, the response to RAI 5.3-12 supplemented DCD, Tier 2, Table 5.3-4, by adding peak vessel fluence and its azimuthal location for 60 calendar years of operation.

The staff reviewed the P/T limits for the ESBWR in accordance with SRP Section 5.3.2 to ensure that adequate safety margins existed for the structural integrity of the ferritic components of the RCPB.

In response to RAI 5.3-6, the applicant stated that it performed the P/T calculation in accordance with the requirements of Appendix G to 10 CFR Part 50. For the representative curves provided, the material initial RT_{NDT} data from the RV specification was used. To calculate the ART (accounting for the effects of irradiation in the vessel beltline region), the copper and nickel specification limits were used in combination with the peak fluence values and the methodology of RG 1.99. This is considered conservative since the actual RT_{NDT} values and chemical composition are normally much lower than the ones specified. Margins for the adjusted reference temperature calculation are consistent with those defined in RG 1.99.

DCD Section 5.3.2.1 states that for each individual component (e.g., main steam nozzle), a finite-element model is used to determine the stresses (pressure and thermal) for the transient events for normal and upset conditions. These stresses are then used to determine the applied K_I for each transient. The most limiting transient K_I for a given pressure and temperature is then compared to the minimum required K_{IC} (note that the minimum temperature limits of Appendix G to 10 CFR Part 50 also apply). The minimum required K_{IC} is based on the limiting RT_{NDT} of the materials for the component (determined as described above) and calculated using the methodology of ASME Code, Section III, Appendix G. For the pressure test condition, a factor of 1.5 is applied to K_{IP} (K_I from primary membrane and bending stresses). For the core-not-critical and core-critical conditions, a factor of 2.0 is applied to K_{IP}. These safety factors are consistent with ASME Code, Section III, Appendix G. Considering that the P-T limits described in the DCD are only representative and that plant specific P-T limits will be provided during the COL application process (see the applicant's response to RAI 5.3-7), the staff finds that the applicants explanation is acceptable.

The staff reviewed the P/T limits imposed based on the ESBWR RV materials to ensure that they meet the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1), as they relate to the selection of materials for the RV and their ability to ensure adequate safety margins for the structural integrity of the RV ferritic components. SRP Section 5.3.2 indicates that P/T limits established for the RCPB must be consistent with the requirements of Appendix G to 10 CFR Part 50 and ASME Code, Section III, Appendix G, to ensure satisfaction of the requirements for RCPB material fracture toughness. The applicant indicated that the temperature limits for core operation (both critical and noncritical), inservice leak tests, and hydrotests are calculated in accordance with Appendix G to 10 CFR Part 50 and ASME Code, Section III, Appendix G. Thus, the probability of the RV material failure and the subsequent effects on reactor core cooling and confinement is minimized. Therefore, the staff finds that the applicant has adequately met the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1).

The staff reviewed the P/T limits imposed on the RV to ensure that the materials selected for the RV meet the relevant requirements of GDC 14, in that they possess adequate fracture toughness properties to resist rapidly propagating failure and to act in a nonbrittle manner. The applicant indicated that the P/T limit curves will be developed in accordance with the criteria of Appendix G to 10 CFR Part 50, thereby ensuring a low probability of significant degradation or gross failure of the RV which could cause a loss of reactor coolant inventory and a reduction in the capability to confine fission products.

The staff reviewed the RV materials to ensure that they meet the relevant requirements of GDC 31 as they relate to behavior in a nonbrittle manner and an extremely low probability of rapidly propagating fracture. In the DCD, the applicant indicated that RG 1.99 is used to calculate the ART. The staff finds this acceptable because RG 1.99 provides methods for predicting the effects of radiation on fracture toughness properties that are applicable to the requirements of GDC 31. In addition, the staff reviewed the P/T limits that will be imposed on the RCPB during preservice hydrostatic tests, inservice leak and hydrostatic tests, heatup and cooldown operations, and core-critical operation. The staff verified that adequate safety margins against nonductile behavior of rapidly propagating failure of ferritic components will exist, as required by GDC 31.

The staff reviewed the RV materials to ensure that they meet the relevant requirements of GDC 32 as they relate to the provision of a materials surveillance program. Compliance with Appendix H to 10 CFR Part 50 satisfies the requirements of GDC 32 for the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine whether they meet the relevant requirements of Appendix H to 10 CFR Part 50, as they relate to determining and monitoring fracture toughness. Section 5.3.2 of this report provides the staff's review of the material surveillance program.

The applicant stated that the P/T limits are consistent with Appendix G to 10 CFR Part 50 and RG 1.99. The applicant also stated that it evaluated the vessel flange, RV head and flange areas, FW nozzles, bottom head, and the core beltline areas. The operating limit curves are based on the most limiting locations. The P/T limits are based on flaw sizes specified in Subsection G-2120 of ASME Code, Section III, Appendix G.

In response to RAI 5.3-7, the applicant stated that a commitment will be provided as a COL action item. This commitment will indicate the need to provide plant-specific information for the development of the P/T limit curves. The applicant revised DCD, Tier 2, Section 5.3.4, to state that the COL applicant will provide the P/T limit methodology in a P/T limits report for NRC review and approval with the plant-specific P/T limits and fracture toughness data.

The applicant confirmed that it performed the fluence analysis using the NRC-accepted methodology documented in NEDC-32983P-A, Class III (Proprietary), and NEDO-32983-A, Class 1 (Nonproprietary), issued December 2001.

As stated above, the applicant provided P/T curves for the ESBWR design which are shown in DCD, Tier 2, Figures 5.3-2 and 5.3-3. The applicant's DCD indicated that these curves are generic curves for the ESBWR RV design. In addition, they are the limiting curves based on the maximum copper and nickel material composition and EOL neutron fluence values. The applicant also indicated that the COL applicant will address the use of plant-specific curves during procurement of the RV. DCD, Tier 2, Section 5.3.4, includes this as a COL action item.

In response to RAI 5.3-8, the applicant stated that the actual RV material properties will be used to refine the P/T curves before plant startup. The data from the surveillance capsules are available after plant startup in accordance with the schedule defined in DCD, Tier 2, Section 5.3.1.6.1. Appendix H to 10 CFR Part 50, which the COL applicant is required to follow, defines the process to be followed if it is necessary to change the P/T curves based on the results of the surveillance program.

As required by 10 CFR 50.34(f)(2)(iii), which references TMI Action Item II.K.3.45, the vessel integrity limits must not be exceeded during rapid depressurization and rapid cooldown. The applicant stated that the ESBWR ADS DPVs are sized such that the vessel depressurization and cooldown are slow enough that vessel integrity limits are not exceeded. The applicant performed a comprehensive thermal analysis which considered the effect of blowdown and the reflooding by the GDCS. Hypothetical ESBWR accidents are much slower than those of currently operating BWRs. In addition, reactor operating procedures are established so that actual transients will not be more severe than those for which the adequacy of the RV design has been demonstrated.

5.3.2.3 Conclusions

DCD, Tier 2, Section 5.3.2, Revision 7 demonstrates that the methodology for fluence calculations adheres to the guidance in RG 1.190, and, therefore, the methodology is acceptable. Fluence calculations require code benchmarking based on plant-specific dosimetry. In this section, the DCD states that the ESBWR design complies with the requirements of Appendix H to 10 CFR Part 50 and ASTM-185, which govern the plant dosimetry requirements. Therefore, the section is acceptable.

The staff also finds that the P/T limits imposed on the RCS for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure conform to the fracture toughness criteria of Appendix G to 10 CFR Part 50. A material surveillance program developed in conformance with Appendix H to 10 CFR Part 50 will determine the change in fracture toughness properties of the RV beltline materials during operation. The use of operating limits, as determined by the criteria defined in SRP Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur. This constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a; Appendix A to 10 CFR Part 50; and GDC 1, 14, 31, and 32.

5.3.3 Reactor Vessel Integrity

The staff reviewed DCD, Tier 2, Section 5.3.3, Revision 7, in accordance with SRP Section 5.3.3, Revision 2. The applicant's assessment of RV integrity is acceptable if it meets codes, standards, and regulatory guidance commensurate with the safety function to be performed. This will ensure that the assessment meets the relevant requirements of 10 CFR 50.55a; Appendices G and H to 10 CFR Part 50; and GDC 1, 4, 14, 30, 31, and 32. These requirements are discussed below:

- GDC 1, GDC 30, and 10 CFR 50.55a(a)(1) require that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

- GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

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

- GDC 31 requires that the RCPB be designed with sufficient margin to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and will minimize the probability of rapidly propagating fracture.

- GDC 32 requires that the RCPB components be designed to permit an appropriate material surveillance program for the RV.

- Appendix G to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviewed the RV materials as they relate to the materials testing and acceptance criteria for fracture toughness contained in Appendix G.

- Appendix H to 10 CFR Part 50 presents the requirements for a material surveillance program to monitor the changes in fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with ASTM E-185. Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the RV. The staff reviewed the RV materials to determine that they meet the relevant requirements of Appendix H as they relate to determining and monitoring fracture toughness.

5.3.3.1 <u>Summary of Technical Information</u>

The RV is a vertical, cylindrical pressure vessel of welded low-alloy steel forging sections. The vessel is designed, fabricated, tested, inspected, and stamped in accordance with ASME Code, Section III, Class 1 requirements. The ESBWR RV dimensions are as follows:

- Nominal inner diameter: 7.112 meters (m) (23.33 ft)
- Nominal wall thickness including clad: 182 mm (7.17 in.)
- Minimum cladding thickness: 3.2 mm (0.125 in.)
- Nominal height from the inside of the bottom head (elevation zero) to the inside of the top head: 27.56 m (90.4 ft)
- Bottom of the active fuel location from elevation zero: 4405 mm (14.45 ft)
- Top of the active fuel location from elevation zero: 7453 mm (24.45 ft)

The cylindrical shell and top and bottom heads of the RV are fabricated of low-alloy steel, the interior of which is clad with stainless steel weld overlays, except for the top head and most nozzles. The main steam and bottom-head drain nozzles are clad with stainless steel weld overlay. The bottom head is clad with nickel-chromium-iron alloy.

A variety of welding processes, such as electroslag, SAW, manual welding, and automated GTAW, are used for cladding, depending on the location and configuration of the item in the vessel. Cladding in the “as-clad” condition may be acceptable for service if deposits are made with automatic processes, such as SAW, GTAW, and electroslag welding. For other processes, particularly where manual welding is employed, some grinding or machining is required. Workmanship samples are prepared for each welding process in the “as-clad” condition and for typically ground surfaces.

The welding material used for cladding in the shell area is ASME Code, SFA-5.9 or SFA-5.4, type 309L or 309MoL, for the first layer, and type 308L or 309L/MoL for subsequent layers. For the bottom-head cladding, the welding material is ASME Code, SFA-5.14, type ERNiCr-3. DCD, Tier 2, Table 5.2-4, lists the materials used in the RV.

The RV is designed and fabricated in accordance with the quality standards set forth in GDC 1 and 30 and 10 CFR 50.55a, as well as the requirements of Section III of the ASME Code. The design and construction of the RV enables inspection in accordance with Section XI of the ASME Code. In addition, the design documents impose additional requirements to ensure the integrity and safety of the RV. Design of the RV and its support system meets seismic Category I equipment requirements.

All plates, forgings, and bolting are 100-percent ultrasonically tested and surface examined by magnetic particle methods or liquid penetrant methods, as required by ASME Code, Section III, Subsection NB. Welds on the RV are examined in accordance with methods prescribed in, and meet the acceptance requirements specified by, ASME Code, Section III, Subsection NB. In addition, the pressure-retaining welds are ultrasonically examined using acceptance standards provided in ASME Code, Section XI.

ISI of the RV must be performed in accordance with the requirements of Section XI of the ASME Code. The RV will be examined once before startup to satisfy the preoperational requirements of Subsection IWB-2000 of the ASME Code, Section XI. Subsequent ISI will be scheduled and performed in accordance with the requirements of 10 CFR 50.55a(g), as described in Section 5.2.4 of this report.

The material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the RV beltline region resulting from exposure to neutron irradiation and thermal environment. Specimens of actual reactor beltline material will be exposed in the RV and periodically withdrawn for impact testing. Operating procedures will be modified in accordance with test results to ensure brittle fracture control.

The RV support is considered a sliding support block type as defined in ASME Code, Section III, Subsection NF-3124. The vessel supports are constructed of low-alloy or carbon steel. Sliding supports are provided at a number of positions around the periphery of the vessel. The vessel support is designed to withstand the loading conditions specified in the design documents and meet the stress criteria of ASME Code, Section III, Subsection NF.

All piping connected to the RV nozzles has been designed not to exceed the allowable loads on any nozzle. Four drain nozzles are provided in the bottom head. Nozzles connecting to stainless steel piping have safe ends or extensions made of stainless steel. These safe ends or extensions are to be welded to the nozzles after the pressure vessel is heat treated to avoid furnace sensitization of the stainless steel. All nozzles, except the drain nozzles and the water-level instrumentation nozzles, are low-alloy steel forgings comprising ASME Code, SA-508, Grade 3, Class 1, material. The safe end materials used are compatible with the material of the mating pipes. The design of the nozzles conforms with ASME Code, Section III, Subsection NB, and meets the applicable requirements of the vessel design documents.

5.3.3.2 <u>Staff Evaluation</u>

Although the staff reviewed most areas separately in accordance with other SRP sections, the importance of the vessel integrity warranted a special summary review of all factors relating to RV integrity. The staff reviewed the fracture toughness of the ferritic materials for the RV, the P/T limits for the operation of the RV, and the materials surveillance program for the RV beltline. SRP Section 5.3.3 provides the acceptance criteria and references that form the bases for this evaluation.

The staff reviewed the information in each area to ensure that inconsistencies did not exist that would reduce the certainty of vessel integrity. The following is a list of the areas reviewed and the sections of this report in which they are discussed:

- RCPB materials (Section 5.2.3)
- RCS pressure boundary ISI and testing (Section 5.2.4)
- RV materials (Section 5.3.1)
- P/T limits (Section 5.3.2)

The integrity of the RV is ensured for the following reasons:

- The RV will be designed and fabricated to the high standards of quality required by the ASME Code and the pertinent ASME Code cases.
- The RV will be fabricated from material of controlled and demonstrated quality.
- The RV will be subjected to extensive PSI and testing to ensure that it will not fail because of material or fabrication deficiencies.
- The RV will operate under conditions, procedures, and protective devices that ensure that the vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients.
- The RV will be subjected to periodic inspection to demonstrate that its high initial quality has not deteriorated significantly under service conditions.
- The RV will be subjected to surveillance to monitor for neutron irradiation damage so that the operating limitations may be adjusted.
- The fracture toughness of the RV materials will be sufficient to ensure that, when stressed under operation, maintenance, testing, and postulated accident conditions, they will behave in a nonbrittle manner and will minimize the probability of rapidly propagating fracture.

The ESBWR RV support is considered to be of a sliding support block type, as defined in ASME Code, Section III, Subsection NF-3124. These supports are not in the region of high neutron fluence, where neutron radiation embrittlement of the supports would be a significant concern. On the basis of the information provided, the staff considers the RV supports for the ESBWR design to be adequately designed to withstand the effects of radiation. Thus, the New Generic Issue 15, "Radiation Effects on Reactor Vessel Supports," is resolved for the ESBWR design.

5.3.3.3 Conclusions

The staff finds that the structural integrity of the ESBWR RV meets the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Appendices G and H to 10 CFR Part 50; and 10 CFR 50.55a. Therefore, the staff finds the structural integrity of the ESBWR RV to be acceptable. The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements of the ESBWR plants conform to the applicable NRC regulations and RGs discussed above, as well as to the rules of Section III of the ASME Code. The ESBWR meets the fracture toughness requirements of the regulations and Section III of the ASME Code, including requirements for surveillance of vessel material properties throughout its service life, in accordance with Appendix H to 10 CFR Part 50. In addition, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G to ASME Code, Section III, and Appendix G to 10 CFR Part 50.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps—Not Applicable to the ESBWR

5.4.2 Steam Generators—Not Applicable to the ESBWR

5.4.3 Reactor Coolant Piping—Not Applicable to the ESBWR

5.4.4 [Reserved]

5.4.5 [Reserved]

5.4.6 Isolation Condenser System

5.4.6.1 Regulatory Criteria

The staff reviewed DCD, Tier 2, Section 5.4.6, Revision 7, using relevant portions of SRP Section 5.4.6, draft Revision 4, “Reactor Core Isolation Cooling System (RCIC),” issued in 1996. Since the ICS is part of the ECCS, the staff also used SRP Section 6.3, Revision 3.

The staff compared the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, GIs, BLs, GLs, or technically significant acceptance criteria beyond those identified in the version used by the staff. Therefore, the staff finds that draft Revision 4 of SRP Section 5.4.6 is acceptable for this review.”

Acceptance criteria are based on the following:

- GDC 4, as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer)

- GDC 5, “Sharing of Structures, Systems, and Components,” as it relates to SSCs important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair the ability of the SSCs to perform their safety function

- GDC 33, “Reactor Coolant Makeup,” as it relates to the system’s capability to provide reactor coolant makeup for protection against small breaks in the RCPB so that fuel design limits are not exceeded

- GDC 34, “Residual Heat Removal,” as it relates to the system design being capable of removing fission product decay heat and other residual heat from the reactor core to preclude fuel damage or RCPB overpressurization

- GDC 54, “Systems Penetrating Containment,” as it relates to the provision of leak detection and isolation capabilities for piping systems penetrating primary containment

- 10 CFR 50.63, “Loss of All Alternating Current Power,” as it relates to design provisions to support the plant’s ability to withstand and recover from a station blackout (SBO) of a specified duration

Section 6.3 of this report presents the acceptance criteria and the evaluation of the ICS as an ECCS.

5.4.6.2 Summary of Technical Information

DCD, Tier 2, Sections 5.4.6.1 and 5.4.6.2, describe the ICS. The ICS removes decay heat after any reactor isolation during power operations. Decay heat removal limits additional pressure rise in the reactor and keeps the RPV pressure below the SRV pressure setpoint. The system consists of four independent loops, each containing a vertical heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the isolation condenser/passive containment cooling system (IC/PCCS) expansion pools, which are vented to the atmosphere.

To place an IC into operation, condensate return valves are opened, whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The ICS can also be initiated manually from the MCR. Each IC has a fail-open nitrogen piston-operated condensate return bypass valve, which opens if the 250-volt direct current power is lost.

The IC/PCCS expansion pool is divided into subcompartments that are interconnected at their lower ends to provide full use of the water inventory for heat removal by any IC. The fuel and auxiliary pools cooling system (FAPCS) performs cooling and cleanup of IC/PCCS expansion pool water. During IC operation, IC/PCCS pool water can boil, and the steam produced is vented to the atmosphere.

ICs are capable of achieving and maintaining safe stable conditions for at least 72 hours without operator action following non-LOCA events. Operator action is credited after 72 hours to refill IC/PCCS pools or initiate SDC.

The IC/PCCS pool has an installed capacity that provides at least 72 hours of reactor decay heat removal capability. The heat rejection process can be continued indefinitely by replenishing the IC/PCCS pool inventory. A safety-related FAPCS makeup line is provided to convey emergency makeup water into the IC/PCCS expansion pool from a water supply outside of the reactor building. The flowpath for this makeup can be established by manually opening the isolation valve on the FAPCS makeup line located at grade level in the yard area external to the reactor building.

The ICS passively removes sensible and core decay heat from the reactor (i.e., natural convection transfers heat from the IC tubes to the surrounding IC/PCCS expansion pool water, and no forced circulation equipment is required) when the normal heat removal system is unavailable.

The ICs are sized to remove postreactor isolation decay heat with three of four ICs operating and to reduce reactor pressure and temperature to safe-shutdown conditions (420 degrees F), with occasional venting of radiolytically generated noncondensable gases to the suppression pool. The ICS operation is independent of station alternating current (ac) power and function whenever normal heat removal systems are unavailable to maintain reactor pressure and temperature below required limits.

The control room operators can perform periodic surveillance testing of the ICS valves by means of remote manual switches that actuate the isolation valves and the condensate return valves. Status lights on the valves verify the opening and closure of the valves. The essential monitored parameters for the IC/PCCS expansion pools are pool water level and pool radiation. IC/PCCS expansion pool water level monitoring is a function of the FAPCS. IC/PCCS expansion pool radiation monitoring is a function of the process radiation monitoring system.

5.4.6.3 Staff Evaluation

The staff assessed the design and function of the ESBWR ICS as described in DCD, Tier 2, Section 5.4.6.

The ICS in the ESBWR is part of the ECCS and also serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main FWS is isolated from the RV. In addition, the ICS will provide decay heat removal necessary for coping with an SBO. The water supply for the ICS pools comes from the condensate demineralizer outlet or from the condensate storage tank.

The ICS is designed and qualified as a safety system. The ICS removes residual and decay heat from the reactor. The system employs natural circulation as the driving head through the IC vertical tubes. The shell side of the condenser is the IC/PCCS expansion pool which boils off to remove heat transferred from the reactor. The heated IC/PCCS expansion pool vents to the outside atmosphere. This is achieved with minimal loss of coolant inventory from the reactor when the normal heat removal system is unavailable subsequent to any of the following events:

- reactor isolation
- SBO
- ATWS
- LOCA

The IC operation automatically limits the reactor pressure and reduces the probability of SRV and ADS operation.

The staff reviewed the diagram to verify that the essential ICS components are designated seismic Category I. The portions of the ICS (including isolation valves) that are located inside the containment and on the steamlines out to the IC flow restrictors are designed to ASME Code, Section III, Class 1, QG A, specifications. Other portions of the ICS are designed to ASME Code, Section III, Class 2, QG B, specifications. The IC/PCCS expansion pools are safety related and seismic Category I. Section 3.6.2 of this report discusses protection of the ECCS against pipe whip and discharging fluids (GDC 4). Section 3.11 of this report discusses environmental qualification of the ECCS equipment.

The ICS consists of four independent loops, and the ICS heat exchangers are sized to remove postreactor isolation decay heat with three out of four ICs operating (101.25 megawatt thermal (MWt)) and to reduce reactor pressure and temperature to safe-shutdown conditions. Since the ICS design is capable of removing fission product decay heat and other residual heat from the reactor core (101.25 MWt), the system meets the requirements of GDC 34.

The IC, connected by piping to the RPV, is placed at an elevation above the RPV. When the steam is condensed, the condensate is returned to the vessel via a condensate return pipe. The steam-side connection between the RPV and the IC is usually open during normal operation. The accumulated subcooled water in the condensate return line is used for reactor coolant makeup during a LOCA.

Any of the following sets of signals generates an actuation signal for the ICS to come into operation:

- two or more MSIV valve positions at less than or equal to 92-percent open, in separate main steam lines, with reactor mode switch in “run” only (percent-open values are those used in the safety analyses)

- RPV dome gauge pressure greater than or equal to 7.447 MPa (1,080 psig) for 10 seconds

- reactor water level below Level 2, with time delay

- reactor water below Level 1

- loss of FW (loss of power to two out of four FW pumps) in reactor run mode

- operator manual initiation

The condensate return line is provided with two parallel valves—an electrohydraulic-operated, main valve which fails as is and a nitrogen piston-operated fail-open valve. This diversity provides more reliability for the system. Two normally closed, fail-closed, solenoid-operated lower head vent valves are located in the vent line from the lower headers. They can be actuated both automatically (when RPV pressure is high and either of the condensate return valves is open) and manually by the control room operator. A bypass line around the lower head vent valves contains one relief valve and one normally closed, fail-open solenoid valve. The valves are designed to open automatically at a pressure setpoint higher than that of the primary lower head vent valves. The vent line from the upper headers is provided with two normally closed, fail-closed, solenoid-operated upper header vent valves to permit opening of the noncondensable gas flowpath by the operator. All of the vent valves will be located in a vertical pipe run near the top of the containment. The vent lines will be sloped to the suppression pool to prevent accumulation of condensate in the piping. During ICS standby operation, discharge of potential entrained hydrogen excess or air is accomplished by a purge line that takes a small stream of gas from the top of the IC and vents it to the MSL. In RAI 5.4-32, S02, the staff asked the applicant for a detailed description of the nitrogen rotary motor-operated valve and the pneumatic piston-operated valve operation, including the actuator. The staff tracked RAI 5.4-32 as an open item in the SER with open items. In the response to RAI 5.4-32 S02, the applicant stated that the ESBWR design will have the option of using either gate valves or ball valves. The steamline isolation valves are nitrogen-powered piston valves and the condensate return valve actuators are electrohydraulic operators, which use an electric motor (a pneumatic motor is used in place of the electric motor) driven pump to drive the piston. RAI 5.4-32 and its associated open item are resolved.

The four radiation monitors in the IC/PCCS expansion pool steam atmospheric exhaust passages for each IC loop are used to detect leakage from the IC outside the containment. Four sets of differential pressure transmitters are located in the steamline and the condensate return line to detect excessive flow as a result of a pipe break or a leak. The IC is isolated automatically when either a high radiation level in the IC pool area is detected or excess flow is detected in the steam supply line or condensate return line.

The IC/PCCS expansion pools have an installed capacity that provides at least 72 hours of reactor decay heat removal capability. The heat rejection can be continued indefinitely by replenishing the IC/PCCS expansion pool inventory. A safety-related independent FAPCS makeup line is provided to convey emergency makeup water into the IC/PCCS expansion pool from the site fire protection system.

The ICS will have controls that will shut down the system if operating conditions exceed certain limits. The ICS is equipped with a leak detection system.

The vendor testing program conducted at PANTHERS/IC test facility in Siet, Italy, included the IC performance test. The Siet test facility also performed full-scale testing on the simplified boiling-water reactor IC. The purpose of the prototype IC test was to show the ability of the IC heat exchanger to meet its design requirements for heat rejection (component performance). Section 21.5 of this report describes the staff evaluation of this vendor test.

Periodic heat removal capability testing of the ICS will be performed. During plant outages, routine ISI is required for the IC, piping, and supports according to ASME Code requirements. The TS provide periodic test and surveillance requirements for essential components of the system. Chapter 14 of this report discusses the proposed initial test program. The ICS is also part of the ECCS, and Section 6.3 of this report describes the evaluation of the ECCS function of the system.

In RAI 5.4-37, the staff identified that the ICS and DPVs are connected to common stub lines from the RV. The staff raised concerns regarding the interaction between the ICS and the DPV when they were located in the same stub tube in the earlier design stage. In the early stages of RCS depressurization, if the ICS is in operation, blowdown through the DPVs may draw fluid back from the cold side of the IC, as well as from the upper part of the RV. Depressurization loads will also affect the ICS, which serves as the primary boundary between the RCS and the environment, since the IC pools are outside of containment. Because the ICS is part of the ECCS, the ICS shall be physically separate from the DPVs, which are also part of the ECCS. The staff requested that the applicant (1) discuss the ramifications of the common tie between the ICS and the DPVs on the stub line from the RV, (2) explain why the design does not meet the physical separation criterion for the ECCS, and (3) describe in detail the potential system interactions and explain why there is no negative impact from the cross-tie between the IC steamline and the DPVs.

In its response, the applicant supplied the following information:

The cross-tie between IC steam line and DPVs in the ESBWR produces no significant negative impact on the loads and safety margins. The key details are as follow:

1. During a LOCA event, the peak operation of ICS occurs during the early part of the depressurization and before the DPV openings.

2. At the time of first DPV opening, there is no sub cooled water inside the IC drain line and in the down comer region. The total dynamic head (DPV flow + IC steam flow) inside the stub tube is small and will not induce backflow into the IC tubes.

3. Failure of one IC drain valve or one DPV valve will not prevent the operation of the other system connecting to the common stub line.

4. Based on (1) and (3), the common-tie between the ICS and DPVs on the stub line has no significant impact on the safety margins (refer to (5) below). Therefore, the physical separation of these two systems is not necessary.

5. Parametric studies were performed with and without the function of the IC heat transfer (i.e., no IC condensation). The results indicate that the long-term containment pressure is slightly higher for the case without the function of IC heat transfer.

The following paragraphs provide additional details on the system interactions as provided in RAI 5.4-37 response:

The nozzles for the stub line and the IC drain line connect to the RPV at elevations of 21.9 and 13.0 m, respectively (reference to the RPV bottom). The bottom of IC tubes is approximately at 6 m above the stub line elevation, or approximately 15 m above the IC drain line nozzle elevation.

In the early stages of RCS depressurization (0 - 500 seconds, before the opening of DPVs), the ICS are in operation and condense significant amount of steam flow (~36 kg/s per IC, MSL break case) from the RPV. The steam flow to the ICS reduces as the RPV pressure decreases and the downcomer water level drops. The first group of ADS valves open after the downcomer level drops below the Level 1.0 set point (11.5 m from the RPV bottom, Table 6.3-1, DCD Rev. 2). Consequently, both the RPV pressure and the steam flow to the ICS reduce further after the first ADS valve opening. The first group of DPV valves opens at 50 seconds after the first ADS valve opening. At this time, the RPV pressure decreases to about 700 kPa (100 psia), the DPV flow is about 7.5 kg/s per DPV and the IC steam flow reduces to about 4 kg/s per IC. The total velocity inside the stub tube is in the range of 35 m/s. The dynamic head is in the range of 2.2 kPa (0.3 psia), which is small compared to the static head of two-phase mixture in the vertical portion of the IC drain line.

At the time of DPV opening, the RPV downcomer as well as the IC drain lines are filled with saturated two-phase mixture due to the fast depressurization resulting

> from the opening of ADS valves. As the result of additional depressurization from the DPV opening, the downcomer two-phase level could swell up a few meters from the Level 1.0 position, and get closer to or below the stub line elevation. However, there is no sub cooled water inside the IC drain line, or inside the downcomer near by the nozzle elevations of the IC drain line or the stub line. In addition, there are loop seals at the lowest elevation of the IC drain lines, near by the injection nozzles. The loop seal provides extra static head, in addition to the 15 meters of static head of the two-phase mixture inside the vertical portion of the IC drain line, to prevent any flow reversal in the IC drain line and steam inlet line due to the DPV opening.

The applicant indicated that the information provided in response to RAI 5.4-37 S01 is included in DCD, Tier 2, Revision 3, Sections 5.4.6.2.2 and 5.4.6.2.3. The staff determined that the information provided in DCD, Tier 2, Revision 3, Sections 5.4.6.2.2 and 5.4.6.2.3 is adequate. Therefore, RAI 5.4-37 is considered resolved.

Section 6.2 of this report discusses containment isolation in accordance with the requirements of GDC 54. GDC 5 is not applicable because the ESBWR is a single-unit plant.

In its response, the applicant stated that the condensate return line is sloped downward from the IC to an elevation below reactor water level to reduce the trapping and collapse of the steam in the drain piping. The staff believes that this sloping will reduce the potential for water hammer events during system startup.

The ICS is designed as a high-pressure reactor coolant makeup system that will start independent of the ac power supply. IC heat exchangers are independent of plant ac power, and they function whenever normal heat removal systems are unavailable to maintain reactor pressure and temperature below limits. Subsequent to an SBO, the system is initiated when the RPV water level 2 is reached following a trip of the FW pumps. The ICS initiates when the condensate return line valve opens using safety-related dc power.

The IC/PCCS expansion pool makeup serves as a clean water supply for replenishing the level during normal plant operation: the FAPCS provides level monitoring.

Because the materials selected for the IC are considered corrosion resistant, leakage across these components to the IC/PCCS pool is not expected. As a result, the prompt identification and response to leakage are important since the leakage indicates degradation of this barrier. In response to RAI 5.4-53, the applicant indicated that the alarm setpoint (in contrast to the high radiation setpoint) is selected close enough to background so that the alarm gives an early warning of a detected leak. In its response, however, the applicant did not indicate the actions to be taken in response to such an IC radiation alarm (e.g., plant shutdown, inspection of the IC tubes), and did it not address why the leak rate associated with a critical size was not used in determining when the IC shall be isolated. The staff tracked RAI 5.4-53 as an open item in the SER with open items.

In RAI 5.4-53 S01, the staff requested that the applicant address the staff concerns in the open item discussed above. The applicant responded stated that effluent monitoring logic initiates an automatic isolation of the affected IC division requiring no immediate operator actions. Followup actions to a radiation detector alarm are directed by the response procedure for the alarm, and operators will carry out actions in accordance with TS-based procedures. Operating plant procedures will address issues such as confirmation of the IC train isolation, investigation and determination of the cause for the isolation, development of a response plan, examination and repair of an IC heat exchanger (if required) or required actions for other equipment, and restoration of the train to operable status in accordance with the TS. With regard to the use of a critical flaw size in determining when the IC shall be isolated, the applicant stated that a critical flaw size is not a key parameter for determining the radiation monitor setpoint in order to maintain the health of any IC train or limit a release from the plant. The automatic IC train isolation at the radiation monitoring system alarm setpoint provides a limit for the rate of release to ensure that the site boundary radiation dose limits are not exceeded. Since the applicant described the actions to be taken in response to a radiation detector alarm as a result of an isolation condenser leak and since the alarm setpoint is based on not exceeding site boundary radiation dose limits, the staff finds that the applicant has adequately addressed the concerns identified in RAI 5.4-53 S01. Therefore, RAI 5.4-53 S01 and associated open item are resolved. Additional information concerning radiation monitor alarm setpoints is located in Section 11.5 of this report.

In RAI 5.4-20, the staff requested that the applicant provide detailed information pertaining to the IC design. In response to RAI 5.4-20, the applicant indicated that the IC tubes would be fabricated from a modified form of Alloy 600 (ASME Code Case N-580-1). However, in other portions of its submittal, the applicant stated that Alloy 600 would be used in the fabrication of the IC tubes. As part of its response to RAI 5.4-20, the applicant also indicated that the IC tubes will be bent by induction bending. However, the applicant did not indicate what effect, if any, this would have on the material properties of the tubing and it did not indicate what testing, if any, was performed to confirm the acceptability of the material properties following bending of the piping and tubing. In RAI 5.4-20 S01, the staff asked the applicant to clarify the actual type of Alloy 600 used in the IC. With regard to the applicant's discussion of induction bending of the IC tubes, the staff requested that the applicant discuss how it has confirmed that the material properties of the most limiting bent tube will remain acceptable following induction bending. The staff also asked the applicant to include a discussion of the material properties tested (e.g., hardness), the results, and the acceptance criteria.

In response to RAI 5.4-20, the applicant indicated that the design of the support structures of the IC tubes was not currently available. The staff notes that, depending on the design, possible crevices between the IC tube and the support could result in the accumulation of chemical contaminants that could lead to corrosion. In addition, the materials of construction of the support are important in that they could corrode and result in a loss of support for, or damage to, the IC tubes. Because material selection and specific design attributes, such as the presence of crevices, can contribute to degradation, the staff requested, in supplemental RAI 5.4-20, that the applicant provide a COL action item to submit this information. The staff tracked RAI 5.4-20, related to the IC as an open item in the SER with open items. Section 6.1.1.3 of this report discusses the resolution of RAI 5.4-20.

Table 6.1-1 indicates that Alloy 600 is used for IC tubing and header fabrication. Alloy 600 has a history of being susceptible to SCC in LWR systems. In RAI 6.1-10, the staff asked the applicant to provide a basis for the use of Alloy 600 in the IC, including material condition (i.e., mill annealed or thermally treated) as it relates to susceptibility to SCC in the reactor coolant and demineralized water environment.

The applicant indicated that there have been no reports of Alloy 600 cracking in BWRs in the absence of a welded crevice or a crack initiated in adjacent Alloy 182. These initiating features are absent from the ESBWR design. In addition, the material used for the IC is the same alloy as that used for the reactor shroud support and stub tubes (see applicant's response to RAI 4.5-18). This alloy (see ASME Code Case N-580-1) is a significantly modified version of Alloy 600, wherein the carbon content is limited, niobium (columbium) is added as a stabilizer, and high-temperature solution heat treatment is required instead of a mill anneal. Stress corrosion resistance is very good. The alloy is approved for use by the ASME Code (Code Case N-580-1) and has been deployed in several operating BWRs, including the Kashiwazaki-Kariwa 6/7 ABWRs. Several of these units have been operating for more than 10 years. In RAI 5.4-55, the staff requested that the applicant discuss the corrosion allowances for Alloy 600 used in the IC. The applicant responded that the Alloy 600 tubing in early BWR ICs performed satisfactorily without incident related to general corrosion in this application. Although general corrosion is not a concern, the applicant did not address whether any other incidences of corrosion or other degradation have occurred in operating units. The staff tracked RAI 5.4-55 as an open item in the SER with open items. Section 6.1.1.3 of this report discusses the resolution of RAI 5.4-55.

For the passive containment cooling system (PCCS), the staff asked the applicant to provide the same information as requested in supplemental RAI 5.4-58. In addition, the staff requested that the applicant discuss whether the cracking that occurred in earlier ICs (refer to applicant's response to RAI 5.4-54) could occur in the PCCS heat exchanger. If so, the staff asked that the applicant discuss the inspections needed to ensure timely detection of cracking. The staff tracked RAI 5.4-57 as an open item in the SER with open items. Section 6.6.3.3 of this report discusses the resolution of RAI 5.4-57.

In response to RAI 5.4-58, the applicant indicated that modified Alloy 600 has been in service for a number of years, but it is not currently inspected as part of a formal ISI program. In RAI 5.2-56, the staff asked the applicant to confirm that the method/technique for the inspection of IC tubes is capable of detecting general wall thinning, pit-like defects, and SCC along the entire length of the tube. In response to RAI 5.4-56, the applicant indicated that, because of the size of the IC tubes (2 nominal pipe size (NPS)), the IC tubes are exempted from volumetric and surface inservice examinations by ASME Code, Section XI, Subsection IWC-1220, which exempts sizes NPS 4 and smaller. The applicant contends that the ICs are subject to leakage (VT-2) examination under ASME Code, Section XI. Given the lack of long-term service experience (with inspection results) and the limitations of accelerated corrosion testing to fully simulate the range of variables that may exist in the field (and are pertinent to corrosion), in supplemental RAI 5.4-58, the staff requested that the applicant provide additional information concerning the inspection and acceptance criteria for the IC tubes or justify why inspection requirements are not needed. The staff noted that the applicant's response to RAI 5.4-56 did not address the information requested by the staff. Therefore, the supplement to RAI 5.4-58 also requested that the applicant address the original issues posed in RAI 5.4-56. The staff tracked RAIs 5.4-56 and 5.4-58 as open items in the SER with open items. Section 6.6.3.3 of this report discusses the resolution of RAIs 5.4-56 and 5.4-58.

In response to RAI 5.4-47, the applicant stated that corrective maintenance for IC tube plugging following tube leak detection can be performed during refueling. After closing the isolation valves to and from the IC and after emptying its pool, personnel operating from the refueling floor can perform subcompartment plugging and repair of the leaking tube. Maintenance will be performed from the upper and lower end, after removal of the header covers. A remotely operated tool will be used to reduce radiation exposure to personnel. If there is considerable damage to some component part of the IC, each module of the IC unit is designed to be easily removable, after cutting the feed, drain, and vent lines. Also, the pool water in a specific IC subcompartment is designed to be removable without requiring the emptying of the remaining IC/PCCS expansion pools.

In response to RAI 5.4-51, the applicant stated that the ICS is designed to remove postreactor isolation decay heat with three out of four IC heat exchangers operating and to reduce the RCS temperature to safe-shutdown conditions of 400 degrees F in 36 hours with occasional venting to the suppression pool of radiolytically generated noncondensable gases. The ICS is capable of achieving and maintaining the safe-shutdown conditions without operator action for at least 72 hours. The safety-related flowpaths of the FAPCS are designed to provide makeup water beyond 72 hours to the ICS. The FAPCS has the ability to supply water to the ICS pools when connected to the fire protection system (FPS). Permanently installed piping is included in the FAPCS, which is connected directly with the site FPS, and this can provide makeup water from 72 hours through 7 days.

Periodic heat removal capability testing of the IC is performed during normal plant operation at 5-year intervals.

On August 10, 2006, the applicant informed the staff that it was incorporating a change to the ICS drainline into the ESBWR design. According to the applicant, the reason for this change is to improve operator flexibility and to maintain minimum chimney collapsed level during a LOCA. This change also reduces the probability of ADS trip in SBO and loss-of-FW events. In addition, this change resulted in the elimination of the Level 1.5 trip and simplified ADS logic with only a Level 1 setpoint. The new inline vessel (tank) is located on each ICS train condensate return line to provide the additional condensate volume for the RPV. The staff requested that the following information shall be added to the ICS ITAAC:

(1) the calculated flow resistance in TRACG between the ICS condensate return line and the reactor

(2) in ITAAC No. 20, the total volume assumed in the analysis for the IC/PCCS expansion pool

The staff requested this change in RAI 14.3-146. This RAI was tracked as an open item in the SER with open items.

In response to RAI 14.3-146, the applicant stated that ICS performance is not determined by controlling the drain line resistance. The heat removal capacity is the key safety significant parameter. The applicant revised the ITAAC, including the heat removal capacity of the ICS and the total volume of the IC/PCCS expansion. Therefore, RAI 14.3-146 and its associated open item are resolved.

In addition, the staff requested, in RAI 5.4-22 S02, that the applicant discuss the means it will use to make certain that the ICS drainline is full during normal operation, thus ensuring that the water volume assumed in the safety analysis is available for injection upon a LOCA signal. The staff tracked RAI 5.4-22 as an open item in the SER with open items. In response to the RAI 5.4-22 S02, the applicant stated that a temperature element will be provided in each condensate return line downstream of the isolation valve and at the bottom and top of the condensate line at the RPV connection. Each temperature will be recorded in the MCR. The temperature measurements can provide information that the condensate line is filled with condensate. The staff was satisfied with the applicant's response. Therefore, RAI 5.4-22 S02 and associated open item are resolved.

Revision 5 deletes T-2, Section 5.4.6-GDC 54, piping systems penetrating containment. The ICS steam supply and condensate lines penetrate the containment. GDC 54 is applicable to the ICS, as indicated in SRP Section 5.4.6, and therefore, is included in the DCD. In response to RAI 5.4.63 S01, the applicant revised DCD Section 5.4.6 to include GDC 54; therefore, this issue is closed.

Section 15.6 of this report discusses the ESBWR design's compliance with 10 CFR 50.63.

5.4.6.4 Conclusions

The NRC staff has reviewed the applicant's information related to the IC system. The staff finds that the applicant has adequately demonstrated that the IC system is capable of decay heat removal during reactor isolation, SBO, and LOCA. The staff finds that the RWCU/SDC system meets the requirements of GDC 4, 5, 33, 34, 64, and 10 CFR 50.63.

5.4.7 Residual Heat Removal

5.4.7.1 Regulatory Criteria

The staff reviewed DCD, Tier 2, Section 5.4.7, Revision7, in accordance with the staff position outlined in the applicable sections of SRP Section 5.4.7, draft Revision 4.

The staff compared the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, GIs, BLs, GLs, or technically significant acceptance criteria beyond those identified in the version used by the staff. Therefore, the staff finds that draft Revision 4 of the SRP Section 5.4.7, is acceptable for this review."

Because of the functional limitations of the passive plant designs, the Commission, in a staff requirements memorandum (SRM) dated June 30, 1994, approved the position in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety systems in Passive Plant Designs," dated March 28, 1994. This position accepts a value of 420 degrees F or lower (rather than the cold shutdown specified in RG 1.139, "Guidance for Residual Heat Removal," issued for comment in May 1978) as the safe stable condition that the passive systems must be capable of achieving and maintaining following non-LOCA events.

Acceptance criteria are based on the following:

- GDC 1, as it relates to the quality standards of the SSCs important to safety

- GDC 2, with respect to the seismic design of the system

- GDC 4, as it relates to the dynamic effects associated with flow instabilities and loads (e.g., water hammer)

- GDC 5, as it relates to SSCs important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair the ability of the SSCs to perform their safety function

- GDC 19, “Control Room,” as it relates to the provision of a control room from which actions can be taken to operate the nuclear power plant

- GDC 34, as it relates to the system design’s capability to remove fission product decay heat and other residual heat from the reactor core to preclude fuel damage or RCPB overpressurization

5.4.7.2 Summary of Technical Information

The SDC mode of the RWCU system is the normal residual heat removal system for the ESBWR. The RWCU/SDC performs the following functions:

- removal of decay heat during normal plant shutdowns

- removal of the core decay heat, plus displacement of the CRD cooling flow approximately 30 minutes following control rod insertion, assuming either the main condenser or ICS is available for initial cool down

- with loss of preferred offsite ac power, bringing the plant to cold shutdown within 36 hours, in conjunction with the ICS, assuming the most restrictive single-active failure

In conjunction with the heat removal capacity of either the main condenser or the ICS or both, the RWCU/SDC system can reduce the RPV pressure and temperature during cooldown operation from the rated design pressure and temperature to below boiling at atmospheric pressure in less than 1 day. The system is also designed to control the reactor temperature reduction rate. The system can be connected to non-safety-related standby ac power (diesel generators), which allows the system to fulfill its reactor cooling functions during conditions when the preferred power is unavailable.

The SDC function of the RWCU/SDC system provides decay heat removal capability at normal reactor operating pressure, as well as at lower reactor pressures. The redundant trains of the RWCU/SDC system permit SDC even if one train is out of service; however, cooldown time is extended when using only one train. If preferred power is lost, the RWCU/SDC system, in conjunction with the ICS, is capable of bringing the RPV to the cold shutdown condition in 36 hours, assuming the most limiting single-active failure, with the ICs removing the initial heat load.

The operation of the RWCU/SDC system at high reactor pressure reduces the plant’s reliance on the main condenser or ICS. The entire cooldown is controlled automatically. As cooldown proceeds and reactor temperatures are reduced, pump speeds are increased and various bypass valves are opened, as described below. During the early phase of shutdown, the RWCU/SDC pumps operate at reduced speed to control the cooldown rate to less than the maximum allowed RPV cooling rate. To maintain less than the maximum allowed RPV cooling rate, both RWCU/SDC trains are placed into operation early during the cooldown, but with the pumps and system configuration aligned to provide a moderate system flow rate. The flow rate for each train is gradually increased as RPV temperature drops. To accomplish this, in each train, the bypass line around the regenerative heat exchanger (RHX) and the bypass line around the demineralizer are opened to obtain the quantity of system flow required for the ending condition of the SDC mode. In addition to the inlet valve to the nonregenerative heat exchanger (NRHX) of the reactor component cooling water system (RCCWS) being open, at an appropriate point, the motor-operated RCCWS inlet valve opens to increase the cooling water flow to each NRHX. The automatic reactor temperature control function governs the adjustable speed drive and controls the cooldown by gradually increasing the speed of the system pumps up to the maximum pump flow. Water purification operation continues without interruption. Over the final part of the cooldown, maximum flow is developed through the RWCU/SDC pumps. Flow rate reduction becomes possible while maintaining reactor coolant temperatures within target temperature ranges. CRD system flow is maintained to provide makeup water for the reactor coolant volume contraction that occurs as the reactor is cooled down. The RWCU/SDC system discharge line is used for fine-level control of the RPV water level as needed.

During hot standby, the RWCU/SDC system may be used as required in conjunction with the main condenser or IC to maintain a nearly constant reactor temperature by processing reactor coolant from the reactor bottom head and the midvessel region of the RV and transferring the decay heat to the RCCWS by operating both RWCU/SDC trains and returning the purified water to the reactor via the FW lines. The pumps and the instrumentation necessary to maintain hot standby conditions can be connected to the standby ac power supply during any loss of preferred power.

The RWCU/SDC system can be used to supplement the FAPCS spent fuel heat removal capacity during refueling (or at other times). The system also can provide additional cooling of the reactor well water when the RPV head is off in preparation for removing spent fuel from the core.

In conjunction with the ICs, the system has the capability of removing the core decay heat, plus drain excess makeup resulting from the CRD purge flow, 30 minutes following control rod insertion.

5.4.7.3 Staff Evaluation

The staff assessed the design and function of the RWCU/SDC for the ESBWR as described in DCD, Tier 2, Section 5.4.7, Revision7.

The ESBWR RWCU/SDC is a non-safety-related system and is not required to operate to mitigate design-basis events. However, some of the valves of the RWCU/SDC perform the following safety-related isolation functions:

- containment isolation of RWCU/SDC lines penetrating containment using containment isolation valves, according to the criteria specified in DCD, Tier 2, Section 6.2.4.
- preservation of the RCS pressure boundary integrity using pressure isolation valves, according to the criteria specified in DCD, Tier 2, Section 5.4.8.

The RWCU/SDC is designed to remove both residual and sensible heat from the core and the RCS during shutdown operations, with the capability to (1) reduce the temperature of the RCS from 270 degrees C (518 degrees F) to 49 degrees C (120 degrees F) within 96 hours after shutdown in conjunction with the heat removal capacity of the main condenser or the ICs or both, and (2) maintain the reactor coolant temperature at 49 degrees C (120 degrees F) for the entire plant shutdown.

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, as well as in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, the staff specified that the ALWR designs shall reduce the possibility of a LOCA outside containment by designing, to the extent practical, all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to full reactor pressure. DCD, Tier 2, Section 5.4.8.1.2, discusses the ESBWR design features that address the intersystem LOCA (ISLOCA). Section 5.4.8.1.2 states that "the supply side of the RWCU/SDC system is designed for the RCPB design pressure plus 10 percent. Downstream of the pumps, the pump shutoff head at 5 percent overspeed is added to the supply side design pressure." The system is designed for operation at reactor pressure; therefore, the ISLOCA issue is resolved for the system.

In SECY-93-087, the staff specified that passive plants must have a reliable means of maintaining decay heat removal capability during all phases of shutdown activities, including refueling and maintenance. The staff's review of the ESBWR design with respect to shutdown operations is based on the applicant's systematic assessment of shutdown operation concerns identified in NUREG-1449, "Shutdown and Low-Power Operations at Commercial Nuclear Power Plants in the United States," issued September 1993. DCD, Tier 2, Section 19.4.7, provides this assessment. Section 19.4 of this report discusses the staff's evaluation of the shutdown operation issues. This section addresses the issues raised in NUREG-1449.

Both adjustable speed drive pumps are connectable to the diesel generator bus during any loss of preferred power supply. There are two redundant trains, and the SDC has the capability to bring the reactor to cold shutdown conditions.

DCD, Tier 2, Section 5.4.8.1.4, describes inspection and testing requirements for the SDC. Preoperational tests, which include valve inspection and testing, flow testing, and verification of heat removal capability, verify the proper operation of the SDC. The inspection and test requirements of the SDC valves are consistent with those identified in DCD, Tier 2, Sections 5.2.4 and 6.2.6, respectively, for the valves that constitute the RCPB and the valves that isolate the line penetrating containment. In addition, DCD, Tier 2, Table 6.2-31, includes these valves, which are subject to IST. The staff finds that the applicant has set proper inspection and test requirements for the SDC valves performing the safety-related functions of containment isolation and RCPB integrity preservation.

The design classifications of the RWCU/SDC components discussed above comply with GDC 1, which specifies that SSCs important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The RCPB portion and the containment isolation valves of the RWCU/SDC are designed as safety Class A. The remaining portions are designed as safety Class B. The system design meets Position C.2 of RG 1.29. This complies with GDC 2, which specifies that the SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes. Section 3.6.2 of this report discusses the protection of the RWCU/SDC system against pipe whip and against discharging fluids (GDC 4). GDC 5 is not applicable to the ESBWR design because the RWCU/SDC system is designed for a single nuclear power unit and is not designed to be shared between units. The RWCU/SDC system is operated from the MCR, thus satisfying the requirements of GDC 19. Because the RWCU/SDC system is not designed to provide safety-related heat removal mitigation of design-basis events, the safety-related ICS complies with the heat removal function of GDC 34.

Safe Shutdown

Establishing a safe-shutdown condition requires maintenance of the reactor in a subcritical condition and adequate cooling to remove residual heat. One of the functional requirements for the ESBWR is that the plant can be brought to a stable condition using the safety-grade systems for all events. The Commission, in an SRM dated June 30, 1994, approved the position proposed in SECY-94-084. This position accepts temperatures of 215.6 degrees C (420 degrees F) or below, rather than the cold shutdown (less than 93.3 degrees C (200 degrees F)) specified in SRP Branch Technical Position RSB 5-1, as the safe, stable condition that the passive decay heat removal system must be capable of achieving and maintaining following non-LOCA events. The SLCS establishes safe shutdown by providing the necessary reactivity control to maintain the core in a subcritical condition and by providing residual heat removal capability to maintain adequate core cooling. DCD, Tier 2, Section 7.4, discusses the systems required for safe shutdown.

For all events, the following systems are used to keep the reactor in a stable condition:

- ICs
- SLCS
- SRVs
- DPVs
- GDCS
- PCCS

The staff finds that the applicant is following the Commission's guidance in SECY-94-084 regarding safe shutdown aspects of the passive plant; therefore, the use of this approach is acceptable.

5.4.7.4 Conclusions

The NRC staff has reviewed the applicant's information related to the RWCU/SDC system. The staff finds that the applicant has adequately demonstrated that the RWCU/SDC system has the capability to cool the RCS following shutdown and provide decay heat removal. The staff further finds that the RWCU/SDC system meets the requirements of GDC 1, 2, 4, 5, and 19.

5.4.8 Reactor Water Cleanup/Shutdown Cooling System

5.4.8.1 Regulatory Criteria

The staff reviewed the RWCU/SDC system description in accordance with SRP Section 5.4.8, Revision 3. Staff acceptance of the design is based on compliance with the requirements of GDC 1, 2, 14, 60, "Control of Releases of Radioactive Materials to the Environment," and 61, "Fuel Storage and Handling and Radioactivity Control." These requirements are discussed below:

- GDC 1, as it relates to the design of the reactor water cleanup system (RWCU) and components to standards commensurate with the importance of the system's safety function

- GDC 2, as it relates to the RWCU being able to withstand the effects of natural phenomena

- GDC 14, as it relates to ensuring the RCPB integrity

- GDC 60, as it relates to the capability of the RWCU to control the release of radioactive effluents to the environment

- GDC 61, as it relates to designing the RWCU with appropriate confinement

RG 1.56, Revision 1, "Maintenance of Water Purity in Boiling Water Reactors," issued July 1978, describes a method acceptable to the NRC staff for implementing the criteria for minimizing the probability of corrosion-induced failure of the RCPB in BWRs by maintaining acceptable purity levels in the reactor coolant. It also describes instrumentation acceptable to the staff for determining the condition of reactor coolant and the coolant purification system.

5.4.8.2 Summary of Technical Information

The two basic functions of the RWCU/SDC system are reactor water cleanup and shutdown cooling (SDC). The RWCU/SDC system performs the reactor water cleanup function during startup, normal power operation, cooldown, and shutdown. The SDC function of the RWCU/SDC system provides decay heat removal capability in conjunction with the main condenser or the ICs at normal reactor operating pressure, as well as at lower reactor pressures. There are two redundant RWCU/SDC trains. Section 5.4.7 of this report presents the review of the SDC function of the RWCU/SDC system.

The RWCU/SDC system consists of the following major components:

- demineralizers
- valves and piping
- RHXs
- NRHXs
- pumps with adjustable speed motor drives

The RWCU/SDC system functions are not safety related; therefore, the system has no safety-related design basis other then to provide a containment isolation function and instrumentation for detection of system breaks outside the containment.

5.4.8.3 Staff Evaluation

The staff reviewed the RWCU/SDC system description in accordance with SRP Section 5.4.8. Staff acceptance of the design is based on compliance with the requirements of (1) GDC 1, as it relates to the design's ability to meet standards commensurate with the system's safety function, (2) GDC 2, as it relates to the system being able to withstand the effects of natural phenomena, (3) GDC 14, as it relates to assuring the integrity of the RCPB, (4) GDC 60, as it relates to the capability of the system to control the release of radioactive effluents to the environment, and (5) GDC 61, as it relates to designing the system with appropriate confinement.

The RWCU/SDC system performs the following functions:

- removes solid and dissolved impurities from the reactor coolant and measures the reactor water conductivity during all modes of operation, in accordance with RG 1.56 and Electric Power Research Institute (EPRI), "BWR VIP-130, BWR Vessel and Internals Project BWR Water Chemistry Guidelines"

- discharges excess reactor water during startup, shutdown, and hot standby conditions and during refueling to the main condenser or to the radwaste system

- minimizes RPV temperature gradients by enhancing circulation through the bottom head region of the RPV and reducing core thermal stratification at low power

- provides containment isolation which ensures that the major portion of the system is outside the RCPB

- provides heated primary coolant for RPV hydrostatic tests and reactor startups

- supplies redundant cleanup capacity with respect to major system components

The RWCU/SDC system is a closed-loop system consisting of two independent pumps and purification equipment. Each train consists of an RHX, an NRHX, a demineralizer, two circulating pumps, isolation valves, piping, and instrumentation. The system takes its suction from the midvessel area of the RPV and from the reactor bottom head and discharges back to the vessel via the FW lines. Water is cooled by flowing through the tube side of the RHX and the NRHX before pump suction. After the NRXH, water moves to the demineralizer to remove all impurities and returns and reheats the reactor water via the shell side of the RHX. Each train is capable of performing the functions of reactor water cleanup and SDC. The system capacity is 1 percent of the rated FW flow rate.

The system is classified as non-safety-related with the exception of the containment isolation valves. The two independent trains are located in the reactor building. System piping from the RPV to the outboard containment isolation valve forms part of the RCPB and is classified as QG A; ASME Code, Section III, Class 1; and seismic Category I. In the remainder of the system, downstream of the containment isolation valves, the piping is classified as QG C;

ASME Code, Section III, Class 3; and seismic Category I. The RWCU/SDC return line from the isolation valve, up to and including the connection to the FW line, is classified as QG B; ASME Code, Section III, Class 2; and seismic Category I.

The staff asked the applicant to provide the basis for designing the return line from the isolation valve, up to and including the connection to the FW line, as QG B. In its response, the applicant stated that the portion of the RWCU/SDC system return line from the isolation valve to the interface with the FW line is designed to QG B to be consistent with the QG of the FW line at the interface. The staff finds the applicant's response acceptable because it is consistent with RG 1.26.

The staff asked the applicant to explain how the effects of high- and moderate-energy piping failures outside the primary containment were evaluated in the RWCU/SDC design to ensure that the other safety-related systems and equipment will not be made inoperable. In its response, the applicant stated that DCD, Tier 2, Sections 3.6.1.2 and 3.6.2.1, describe protection against dynamic effects associated with postulated rupture of piping outside the containment for high- and moderate-energy piping. The description includes the identification of the high-energy piping located outside the containment, the potential damage resulting from dynamic effects, the design-basis compartment break, compartment pressurization, and equipment qualification. Sections 3.6.1 and 3.6.2 of this report present the staff evaluation of these DCD sections.

In the same letter, the staff asked the applicant to demonstrate the capability of safety-related systems to withstand the effects of postulated internally generated missiles from the RWCU/SDC system both inside and outside the primary containment. In its response, the applicant stated that DCD, Tier 2, Section 3.5.1, includes the evaluation of the ability of the safety-related systems to withstand the effects of internally generated missiles both inside and outside containment. Section 3.5.1 of this report discusses the staff's evaluation of this DCD section. In addition, the staff asked the applicant to demonstrate the capability of structures housing the RWCU/SDC, including safety-related components and instruments inside these structures, to withstand external and internal flood conditions. In its response, the applicant stated that the RWCU/SDC system components are housed in the containment and the reactor building. DCD Section 3.4 describes the internal and external flooding evaluation. As a result of its response to RAI 5.4.10, the applicant revised DCD, Tier 2, Sections 3.4.1.3, 3.4.1.4, and 3.4.1.4.2, to further clarify its flooding analysis. The staff found the applicant's response acceptable and confirmed the changes in DCD, Tier 2, Revision 2. Section 3.4.1 of this report presents the staff's evaluation of this DCD section.

Based on this seismic and QG classification design information, the staff finds that, by following the guidelines of RGs 1.26 and 1.29, the applicant has met the requirements of GDC 1 and 2 as they relate to the ability of the RWCU/SDC design to meet standards commensurate with the system's safety function and to withstand the effects of natural phenomena.

The two safety-related containment isolation valves on the suction lines of the RWCU/SDC system receive isolation signals from the leak detection and isolation system. These valves will automatically isolate on the following indications:

- high RWCU/SDC flow
- low reactor water level (Level 2)
- high temperature in the MSL tunnel

- initiation of the SLCS

The suction lines of each train are isolated by one automatic nitrogen-operated gate valve inside and one air-operated gate valve outside the containment. The reactor bottom suction line has a sampling line isolated by one automatic nitrogen-operated globe valve inside and one air-operated globe valve outside the containment. RWCU/SDC pumps, heat exchangers, and demineralizers are located outside the containment. In addition, the applicant's DCD states that the RWCU/SDC meets the requirements of RG 1.56 and the EPRI report entitled, "BWR VIP-130, BWR Vessel and Internals Project BWR Water Chemistry Guidelines." Based on this information, supplemented by the seismic and QG classifications for the portion of the system including the containment isolation valves discussed above, the staff finds that the system meets the requirements of GDC 14 as it relates to assuring the integrity of the RCPB.

The staff asked the applicant to describe the design features of the RWCU/SDC system that will control the release of radioactive effluents to the environment in accordance with GDC 60. In its response, the applicant stated that contaminated liquid waste will be transferred to the liquid waste management system (LWMS). In addition, flushing connections are provided to decontaminate piping and equipment such as the demineralizers and the heat exchangers. The RWCU/SDC system is provided with piping connections routed to the main condenser and the LWMS. The piping has butt-welded connections, rather than socket welds, to reduce crud traps. If high radiation is detected downstream of the demineralizer, the flow will be manually shifted to the LWMS by first opening the remote manual isolation valve to the LWMS and then closing the remote manual system isolation valve to the main condenser. The staff finds the applicant's response acceptable because the purpose of the LWMS is to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including AOOs. Based on this information, the staff finds that the system meets the requirements of GDC 60 as it relates to the capability of the RWCU to control the release of radioactive effluents to the environment.

The demineralizers used are of the mixed-bed type with nonregeneration-type resin beads. A resin strainer capable of removing resin particles from the demineralizers' effluent is located at the outlet line to prevent resin beads from entering the system. Resin bed performance is monitored by the process sampling system. Sample probes are located in the inlet and outlet lines of the two demineralizers where samples are routed to the sample station for analysis. In addition, the conductivity of the demineralizer influent and effluent streams is continuously measured and transmitted to the MCR. The performance of the resin beads determines their replacement time. Since nonregeneration-type resin beads are used, whenever it is necessary to replace the spent resins, the resin vessel will be isolated from the rest of the system before resin addition. The staff asked the applicant to describe the resin transfer system and indicate the provisions taken to ensure that transfers are complete and that crud traps in transfer lines are eliminated. In its response, the applicant stated that the details of the resin transfer system will be designed in the detail design phase and that it will add the following design description in DCD, Tier 2, Section 5.4.8.1.2:

> The resin transfer system will be designed to prevent resin traps in sluice lines. Consideration will be given in the design to avoid collection of resins in valves, low points and stagnant areas.

The applicant committed to placing this statement in a future revision of the DCD. The staff found the applicant's response acceptable and confirmed the changes in DCD Revision 5.

Spent resins will be sluiced to a backwash-receiving tank from which they will be transferred to the radwaste system for processing and disposal. Demineralizers are located in separate concrete-shielded cubicles that are accessible through shielded hatches. Valves and piping within the cubicles are reduced to the extent that entry into the cubicles is not required during any operational phase. Most of the valves and piping are located in a shielded valve gallery adjacent to the demineralizer cubicles. The valves are remotely operable to the greatest practical extent to minimize entry requirements into this area. The backwash tank is shielded separately from the resin transfer pump.

Each demineralizer is protected from high flow, high differential pressure across the strainer and across the demineralizer and from demineralizer inlet high temperature by a bypass valve. In the event of high differential pressure or high temperature, an alarm will be activated in the MCR to alert the plant operator and will automatically isolate the demineralizer by first opening the bypass valve and then closing the demineralizer inlet valve.

The staff asked the applicant to describe the control features that will prevent inadvertent opening of the demineralizer backwash valves during normal operation. In its response, the applicant stated that interlocks are provided to prevent inadvertent opening of the resin addition and back-flushing valves during normal operation. The staff finds the applicant's response acceptable because the use of interlocks will be adequate to prevent the inadvertent opening of the valves.

SRP Section 5.4.8 states that, to prevent resin loss from the demineralizer bed, the RWCU system shall include a means for automatically maintaining flow through demineralizer beds in the event of low-process flow or loss of flow. The staff asked the applicant to describe design requirements for a system controlling the ability of the demineralizer to automatically adjust flow through its resin beds to prevent resin loss in the event of a decrease of system flow. In its response, the applicant stated that this SRP requirement does not apply to the ESBWR demineralizers because they use nonregeneration bead-type resins which do not lose resins on a reduction or loss of process flow. The staff finds the applicant's response acceptable.

The RHX and the NRHX are other components of the RWCU/SDC system that are exposed to high-radiation levels. These components are also located in shielded cubicles with valves operated remotely by use of extension valve stems or from instrument panels located outside the cubicle.

The cleanup flow leaving the NRHX and going into the demineralizers shall be of a specific temperature; therefore, the NRHX shall be able to maintain the required temperature of the cleanup flow when its cooling capacity is reduced as a result of partially bypassing a portion of the return flow to the main condenser or the radwaste system. The staff asked the applicant to describe whether the NRHX has the capacity of maintaining the desired temperature when its return flow is reduced. In its response, the applicant stated that the NRHX performance was evaluated in the cleanup mode with a reduced RHX capacity by assuming that 25 percent of its normal return flow is bypassed to the main condenser. Since the NRHX cools the reactor water by transferring heat to the RCCWS, increasing the water flow of the RCCWS will provide enough cooling capacity to maintain the required temperature of the cleanup flow to the demineralizer. This proved sufficient to maintain the demineralizer's required inlet temperature. The staff finds the applicant's response acceptable.

Based on this information, the staff finds that the RWCU/SDC system design meets the requirements of GDC 61 as it relates to designing the system with adequate confinement features.

5.4.8.4 Conclusions

The RWCU/SDC system will be used to maintain the reactor water purity and to reduce the reactor water inventory as required by plant operations. The staff's review has included system schematics along with descriptive information concerning the system design and operation.

The staff finds that the proposed design of the RWCU/SDC system is acceptable and meets the relevant requirements of GDC 1, 2, 14, 60, and 61. This conclusion is based on the following:

- The applicant has met the requirements of GDC 1 by designing, in accordance with the guidelines of RG 1.26, the portion of the RWCU/SDC extending from the RV and recirculation loops to the outermost primary containment isolation valves to QG A and by designing, in accordance with Position C.2 of RG 1.26, the remainder of the system outside the primary containment to QG C.

- The applicant has met the requirements of GDC 2 by designing, in accordance with Positions C.1, C.2, C.3, and C.4 of RG 1.29, the portion of the RWCU/SDC extending from the RV and recirculation loops to the outermost primary containment isolation valves to seismic Category I.

- The applicant has met the requirements of GDC 14 by meeting the positions of RG 1.56 in maintaining reactor water purity and material compatibility to reduce corrosion probabilities, thus reducing the probability of RCPB failure.

- The applicant has met the requirements of GDC 60 and 61 by designing a system containing radioactivity with confinement and by venting and collecting drainage from the RWCU/SDC components through closed systems.

Based on this information, the staff finds that the RWCU/SDC design for the ESBWR is acceptable.

5.4.9 Main Steamlines and Feedwater Piping

The applicant provided information regarding MSL and FW piping in DCD, Tier 2, Section 5.4.9. Chapter 10 of this report presents the staff's evaluation of these systems.

5.4.10 Pressurizer—Not Applicable to the ESBWR

5.4.11 Pressurizer Relief Discharge System—Not Applicable to the ESBWR

5.4.12 Reactor Coolant High-Point Vents

5.4.12.1 Regulatory Criteria

The staff performed its review of the ESBWR RCS high-point vent system in accordance with SRP Section 5.4.12, draft Revision 1, issued in 1996.

The staff compared the SRP version used during the review with the 2007 version of the SRP. The 2007 version did not include any requirements, GIs, BLs, GLs, or technically significant acceptance criteria beyond those identified in the version used by the staff. Therefore, the staff finds that SRP, draft Revision 1, Section 5.4.12, is acceptable for this review."

The following requirements appear in 10 CFR 50.34(f)(2)(VI):

> Provide the capability of high point venting of non-condensable gases from the RCS, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of LOCA or an unacceptable challenge to containment integrity (II.B.1).

Acceptance criteria are based on the following:

- 10 CFR Part 50.55a and GDC 1 and 30, as they relate to the vent system components that are part of the RCPB being designed, fabricated, erected, and tested and maintained to high quality standards
- GDC 14, as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture
- 10 CFR 50.46(b), as it relates to the long-term cooling of the core following any calculated successful initial operation of the ECCS to remove decay heat for an extended period of time
- 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," with respect to environmental qualification of electrical equipment necessary to operate the reactor coolant vent system
- GDC 17, "Electric Power Systems," with respect to the provision of normal and emergency power for the vent system components
- GDC 19, with respect to the vent system controls being operable from the control room
- GDC 36, as it relates to the vent system being designed to permit periodic inspection

5.4.12.2 Technical Information

The ESBWR has an RPV head vent system that handles any noncondensable gas buildup at the high point inside the RPV head by sweeping the gases through an MSL and then ultimately to the condenser. Additionally, systems that are connected to the RPV and are stagnant during normal plant operation have lines that are sloped to prevent any buildup of noncondensable gases.

During reactor operation, the noncondensable gases that may collect in the reactor head and the IC steamlines are drawn to the steamline through a vent line with two normally open motor-operated valves from the RPV head to the MSL and a purge line from each of the ICs. Differential pressure between the reactor head and the downstream steamline location extracts the noncondensables. The noncondensables are swept from these lines to the condenser, where they are extracted. These vents and purge lines are not required to ensure natural circulation core cooling. The vent line used to vent the reactor head noncondensables following a refueling operation is isolated with two normally closed valves during reactor power operation. The ICs also vent noncondensables to the suppression pool to maintain their performance; however, the ICs are isolable and not part of the primary system. Section 5.4.6 of this report discusses the IC vents.

5.4.12.3 <u>Staff Evaluation</u>

The staff reviewed the design and function of the RPV vent system, as described in DCD, Tier 2, Section 5.4.12, Revision 7.

The ESBWR meets the requirements of 10 CFR 50.34(f)(2)(VI), which references TMI Action Item II.B.1, regarding the capability of high-point venting of noncondensable gases from the RCS. The noncondensables are swept from the steamlines to the condenser, where they are extracted. Position indication and controls for opening and closing the valves are in the control room. These vents and purge lines are not required to ensure natural circulation core cooling. The staff reviewed the procedure for operation of the RPV head vent system information provided in DCD, Tier 2, Section 5.4.12.1 and found it to be acceptable.

When the RPV is in an isolated condition, the RPV head vent line and the SRVs provide redundancy for venting the RCS. The vent line used to vent the reactor head noncondensables following a refueling operation is isolated with two normally closed valves during reactor power operation. These valves are subject to an environmental qualification (10 CFR 50.49(a)) program, as described in DCD, Tier 2, Section 3.11 and evaluated in Section 3.11 of this report.

GDC 17 is met by an onsite electric power system that provides normal and emergency power to permit operation of the RPV head vent line valves. GDC 19 is met by controls and indication that permit operation of the valves from the MCR. The RPV head vent system is not part of the ECCS and is not required to ensure natural circulation core cooling. Therefore, GDC 36 does not apply. For RCPB isolation purposes during reactor power operation, the use of two nitrogen-operated valves in series in the piping that vents the RPV to the equipment and floor drain sump provides redundancy. Either or both valves isolate the piping. Failure modes consist of loss of power supply, failure of the control system, and mechanical failure in the valve. If one of the valves experiences a failure, the second valve in series performs the isolation function. Indication of open and closed position and temperature downstream of the second valve are available to operators in the control room.

A connection at the RPV flange area links the internal integral head vent piping to the external head vent piping. The piping is 2 inches in diameter. The vent piping directs air and noncondensable gases from the RPV to either the equipment and floor drain sump or one of the MSLs. The vent piping permits air to be released from the RPV so that the vessel can be filled with water for hydrostatic testing, vents gases during reactor operation and reactor shutdown, and provides the upper tap for RPV-level measurement during reactor shutdown. The diameter of the vent line piping is much smaller than the diameter of the MSL piping. Therefore, a break in this piping is bounded (in accordance with 10 CFR 50.46a, “Acceptance Criteria for Reactor

Coolant System Venting Systems”) by an MSL break, which is addressed in DCD, Tier 2, Section 6.3.

5.4.12.4 Conclusions

As discussed above, the RPV vent system design for the ESBWR complies with the guidelines of SRP Section 5.4.12 and therefore is acceptable. The staff finds that the design of the RCS high-point vents is acceptable because it meets the relevant requirements of 10 CFR 50.34(f)(2)(vi), 10 CFR 50.46a, 10 CFR 50.49, 10 CFR 50.55a, TMI-2 Action item II.B.1, and GDC 1, 14, 17, and 19. The staff finds that the ESBWR design provides various means to prevent accumulation of noncondensable gases in the RCS.