5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

In this section of the Final Safety Analysis Report (FSAR), the applicant identifies various changes that are included in the following sections of Chapter 5. The identified changes have been evaluated in various subsections below. Therefore, this section does not require additional U.S Nuclear Regulatory Commission (NRC) staff technical evaluation.

Section 5.1 of the South Texas Project (STP) Units 3 and 4 combined license (COL) FSAR incorporates by reference Section 5.1 "Summary Description," of the advanced boiling-water reactor (ABWR) design control document (DCD), Revision 4, referenced in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review[1]. The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the summary description have been resolved.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1.1 *Compliance with 10 CFR Part 50, Section 50.55 [Related to RG 1.206, Section 5.2.1.1, "Compliance with 10 CFR 50.55a"]*

5.2.1.1.1 Introduction

This section addresses the use of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) and the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), consistent with the requirements of 10 CFR 50.55a used by the STP Units 3 and 4 FSAR.

5.2.1.1.2 Summary of Application

Subsection 5.2.1.1 of the STP Units 3 and 4 COL FSAR incorporates by reference Subsection 5.2.1.1 of the ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements.

5.2.1.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling-Water Reactor Design," (July 1994) (FSER related to the ABWR DCD).

In addition, the relevant requirements of the Commission regulations for the Integrity of Reactor Coolant Pressure Boundary, and the associated acceptance criteria, are in Section 5.2 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (the Standard Review Plan [SRP]).

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[1] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

The NRC regulations in 10 CFR Parts 50 and 52 provide the regulatory basis for the NRC staff to review the information in the STP COL application. For example, NRC regulations in 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, "Quality standards and records," require nuclear power plant structures, systems, and components (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. Furthermore, the NRC regulations in 10 CFR 50.55a, as they relate to the establishment of minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, require conformance with the appropriate editions of published industry codes and standards.

NRC staff followed the guidance in SRP Subsection 5.2.1.1, "Compliance with the Codes and Standards Rule 10 CFR 50.55a," and Regulatory Guide (RG) 1.206 in evaluating STP FSAR Subsection 5.2.1.1 for compliance with the NRC regulations.

5.2.1.1.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Subsection 5.2.1.1 of the certified ABWR DCD. The staff reviewed Subsection 5.2.1.1 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[2] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the compliance with the 10 CFR 50.55a.

NUREG–1503 states that all ASME Code Class 1, 2, and 3 pressure-retaining components and their supports shall be designed in accordance with the requirements of ASME Code Section III using the specific edition and addenda in the ABWR DCD. This NUREG also states that 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. The COL applicant must identify in the application the portions of later code editions and addenda for the staff to review and approve.

The staff reviewed STP FSAR Subsection 5.2.1.1 to evaluate its compliance with the requirements in 10 CFR Parts 50 and 52. The staff confirmed that the information in the application and the information incorporated by reference address the relevant information related to codes and standards.

ABWR DCD Tier 2 Subsection 5.2.1.1 refers to Table 3.2-3, "Quality Group Designations – Codes and Industry Standards," for the ASME Code applied to plant components and states that the Code Edition, applicable addenda, and component dates will be in accordance with 10 CFR 50.55a. The staff issued Request for Additional Information (**RAI) 05.02.01.01-1** requesting the applicant to specify the Code Edition and Addenda to be applied to STP components. The RAI also requested the COL applicant to specify the edition and addenda of the ASME OM Code to be applied to pumps, valves, and dynamic restraints at STP Units 3 and 4. The applicant's response to RAI 05.02.01.01-1 dated September 14, 2009 (ML092580477) refers to the response to RAI 03.02.02-5 submitted in a letter dated

2 See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

August 26, 2009 (ML092430131). According to the applicant, the Code Edition and Addenda to be applied to the STP components are listed in ABWR DCD Tier 2 Table 1.8-21, “Industrial Codes and Standards Applicable to ABWR”; in STP COL FSAR Table 1.8-21 of the same title; and in Table 1.8-21a, “Codes and Standards for Site-Specific Systems.” The COL applicant’s response to RAI 03.09.06-2 dated August 17, 2009 (ML092310488), states that STP COL FSAR Table 1.8-21a will be revised to include the 2004 Edition of the ASME OM Code. The staff found that the 2004 Edition of the ASME OM Code is incorporated by reference in 10 CFR 50.55a of the NRC regulations, and it is therefore sufficient as part of the description of the IST program in support of the STP Units 3 and 4 COL application. As specified in 10 CFR 50.55a, the IST Program for the initial 120-month interval for a licensee under 10 CFR Part 52 must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a, on the date 12 months before the date scheduled for initial fuel loading (or the optional ASME Code Cases listed in the applicable RG incorporated by reference in 10 CFR 50.55a), subject to the limitations and modifications listed in 10 CFR 50.55a. This RAI was tracked as **Confirmatory Item 05.02.01.01-1**. Subsequently, the applicant submitted Revision 4 to STP COL FSAR Table 1.8-21a that includes the 2004 Edition of the ASME OM Code. Therefore, **Confirmatory Item 05.02.01.01-1** is closed and **RAI 05.02.01.01-1** is resolved.

5.2.1.1.5 Post Combined License Activities

There are no post COL activities related to this subsection.

5.2.1.1.6 Conclusion

The NRC staff’s finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff’s review confirmed that the applicant has addressed the required information relating to the compliance with 10 CFR 50.55a. No outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to compliance with 10 CFR 50.55a that were incorporated by reference have been resolved.

The staff’s review confirmed that STP FSAR Subsection 5.2.1.1 adequately incorporates by reference ABWR DCD Subsection 5.2.1.1. Therefore, the staff concluded that the information in STP FSAR Subsection 5.2.1.1 satisfies the NRC requirements in 10 CFR Parts 50 and 52.

5.2.1.2 *Applicable Code Cases [Related to RG 1.206, Subsection 5.2.1.2, “Compliance with Applicable ASME Code Cases”]*

5.2.1.2.1 *Introduction*

This SER subsection addresses the ASME Code cases used to provide assurance for the integrity of the reactor coolant pressure boundary (RCPB) at the STP Units 3 and 4. This section also discusses the applicable NRC RGs that indicate the acceptance of ASME Code cases with or without conditions.

5.2.1.2.2 *Summary of Application*

Section 5.2.1.2 of the STP Units 3 and 4 COL FSAR incorporates by reference Subsection 5.2.1.2 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 5.2, the applicant provides the following:

Tier 2 Departure Requiring NRC Approval*

- STD DEP 1.8-1 — Tier 2* Codes, Standards, and Regulatory Guide Edition Changes

This departure provides the applicable Tier 2* Codes, Standards, and RG Edition Changes identified in the ABWR DCD in STP FSAR Tables 5.2-1 and 5.2-1a. This departure states that currently approved ASME Code cases per RG 1.84 (Revision 33), "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," may be used in the future. STD DEP 1.8-1 also notes that RG 1.85, "Materials Code Case Acceptability, ASME Section III, Division 1," is obsolete and has been deleted. RG 1.85 is now incorporated into RG 1.84 (Revision 33).

5.2.1.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1503. In addition, the relevant requirements of the Commission regulations for the applicable Code cases, and the associated acceptance criteria, are listed in Subsection 5.2.1.2 of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 2* departures. Tier 2* departures require prior NRC approval and are subject to the requirements of 10 CFR Part 52 Appendix A, Section VIII.B.6.

NRC regulations in 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records," require nuclear power plant 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. Furthermore, the NRC regulations in 10 CFR 50.55a related to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components require conformance to the appropriate editions of published industry codes and standards.

As one acceptable means of meeting the applicable NRC regulations, RG 1.84 lists Code cases related to Section III, "Rules for Construction of Nuclear Facility Components," in the ASME BPV Code that are acceptable with applicable conditions for the design, fabrication, materials, and testing of components at nuclear power plants. RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," lists Code cases related to Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," in the ASME BPV Code that are acceptable with applicable conditions for use in the inservice inspection (ISI) of nuclear power plant components and their supports. RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," lists acceptable Code Cases related to the ASME OM Code that are oriented to the operation and maintenance of nuclear power plant components, with the applicable conditions for implementation at nuclear power plants.

NRC staff followed the guidance in NUREG–0800 Subsection 5.2.1.2, “Applicable Code Cases,” and RG 1.206 in evaluating STP FSAR Subsection 5.2.1.2 for compliance with the NRC regulations.

5.2.1.2.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Subsection 5.2.1.2 of the certified ABWR DCD. The staff reviewed Section 5.2.1.2 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[3] The staff’s review confirmed that the information in the application and the information incorporated by reference address the required information relating to the compliance with the ASME Code cases.

Tier 2 Departure*

- STD DEP 1.8-1 Tier 2* Codes, Standards, and Regulatory Guide Edition Changes

In NUREG–1503, NRC staff described the evaluation of the ASME Code cases identified in the ABWR DCD for use by a COL applicant implementing the ABWR reactor design. The staff concluded that the ASME Code cases identified in ABWR DCD Tier 2 Table 5.2-1, “Reactor Coolant Pressure Boundary Components Applicable Code Cases,” are acceptable as specified in the applicable NRC regulatory guides or have been reviewed and found to be acceptable by the staff for use in the ABWR design. The staff found that compliance with the requirements of these Code cases will result in the quality of a component commensurate with the importance of the safety functions of the components that satisfy the requirements of GDC 1. The staff stated that a COL applicant may identify in the COL application the intent to use additional Code cases provided that they do not alter the staff’s safety findings on the ABWR certified design.

ABWR DCD Tier 2 Section 5.2.1.2 states that RG 1.84, RG 1.85, and RG 1.147 provide a list of ASME Design and Fabrication Code cases that have been generically approved by the staff, and that Code cases on this list may be used for design purposes until they are appropriately annulled. In STD DEP 1.8-1, the STP COL applicant notes that RG 1.85 is incorporated in RG 1.84. The staff issued **RAI 05.02.01.02-1** requesting the STP COL applicant to indicate the Code Cases listed in RGs 1.84 and 1.147 that the applicant plans to use at STP Units 3 and 4. The applicant’s response to this RAI dated September 24, 2009 (ML092710223), states that FSAR Tier 2 Subsection 6.6.9.1, “PSI and ISI Program Plan,” discusses the applicable ASME BPV Code Section XI Edition and Addenda, as well as an outline and schedule for the plant-specific preservice inspection (PSI) and ISI Program Plan. STP FSAR Tier 2 Subsection 3.9.7.3, “Pump and Valve Testing Program,” addresses the optional implementation of the ASME Code cases listed in RG 1.147. STP FSAR Tier 2 Table 5.2-1 lists the Code Cases of RG 1.147 that are applicable to the pressure-retaining ASME BPV Code Section III Class 1, 2, and 3 components. This RAI was tracked as **Open Item 05.02.01.02-1** in the SER with open items. The NRC staff is reviewing the description of the ISI Program for STP Units 3 and 4 in other sections of this SER. Therefore, Open Item 05.02.01.02-1 is resolved.

[3] See *“Finality of Referenced NRC Approvals”* in SER Section 1.1.3, for a discussion on the staff’s review related to verification of the scope of information to be included in a COL application that references a design certification.

ABWR DCD Tier 2 Section 5.2.1.2 and Table 5.2-1 do not address ASME OM Code Cases to be applied in ABWR plants. The staff issued **RAI 05.02.01.02-2** requesting the STP COL applicant to specify the ASME Code cases to be applied at STP Units 3 and 4, including the acceptance of the Code cases in RG 1.192 and any conditions specified in RG 1.192, as part of fully describing the IST Program. The applicant's response to this RAI in a letter dated September 24, 2009, states that the applicant is applying the ASME OM Code, 2004 Edition, to the STP Units 3 and 4 IST Program. This RAI was tracked as **Open Item 05.02.01.02-2**. The staff is reviewing the description of IST Program for STP Units 3 and 4, including the planned use of ASME OM Code cases in STP FSAR Section 3.9.6. The staff will complete its review of the IST Program description, as discussed in Section 3.9.6 of this SER. Therefore, **Open Item 05.02.01.02-2** is resolved.

ABWR DCD Tier 2 Subsection 5.2.1.2 states that annulled cases are considered active for equipment that was contractually committed for fabrication before the annulment. The staff issued **RAI 05.02.01.02-03** requesting the STP COL applicant to discuss plans to implement this provision in the ABWR DCD regarding the use of annulled Code cases. The COL applicant's response to this RAI in a letter dated September 24, 2009, states that ASME Code requirements will be specified as part of the purchase orders for equipment, including applicable ASME Code cases. The Code Cases to be applied to that equipment will be "frozen" at the time the purchase order is accepted by the supplier. Any ASME Code Cases that are annulled by ASME after that time may still be considered active, with respect to equipment where the supplier has already accepted a purchase order. The COL applicant also indicates that every reasonable effort will be made not to apply annulled Code Cases to the equipment, if at all possible, even after the release of the purchase order. The applicant and the STP equipment suppliers will consider the reasons for ASME Code case annulments to ensure that there are no detrimental impacts on the equipment or its function. The staff found that the COL applicant has clarified the implementation of the ABWR DCD provision regarding the use of annulled Code cases. The plans the COL applicant describes provide reasonable assurance that the use of Code Cases that are annulled after issuing a purchase specification will be evaluated to ensure that there are no detrimental impacts on the equipment or its function. As part of the inspection activities, the staff will also have an opportunity to review the use of any annulled Code Cases. The staff found the applicant's response acceptable, and **RAI 05.02.01.02-3** is therefore resolved.

STP FSAR Section 1.8, "Conformance with Standard Review Plan and Applicability of Codes and Standards," states that the STP FSAR conforms to RG 1.84 Revision 33, which is incorporated by reference in 10 CFR 50.55a. STP FSAR Table 5.2-1 specifies several changes to the list of Code Cases in the ABWR DCD. The applicant has added Code Case N-71-17. However, the most recent version of Code Case N-71 listed in RG 1.84 (Revision 33) is N-71-18. In addition, Table 5.2-1 lists Code Case N-71-15, which was superseded by Code Case N 71-18, as discussed above. Table 5.2-1 identifies Code Case N-319-3 as listed in RG 1.84 Revision 33. However, the applicant also lists superseded Code Case N-319. Code Case N-580-2 was added to Table 5.2-1. However, this code case has not been endorsed by the staff in RG 1.84. Code Case N-580-1 is listed in RG 1.84 Revision 33. The staff issued **RAI 05.02.01.02-4** requesting the applicant to address these concerns.

The applicant's response to RAI 05.02.01.02-4 dated September 24, 2009, states that Table 5.2-1 will be modified to reference Code Case N-71-18 in lieu of Code Case N-71-17. The staff found this response acceptable because the applicant will use the most recent revision of Code Case N-71 listed in RG 1.84, Revision 33. As a result, Revision 4 to STP FSAR Table 5.2-1 references Code Case N-71-18. Therefore, **RAI 05.02.01.02-4** is resolved.

The applicant states that the use of Code Case N-580-2 is discussed in the applicant's response to RAI 04.05.02-3. The staff noted that Code Case N-580-1 is the most recent revision of this Code case listed in RG 1.84. The applicant's response to RAI 04.05.02-3 states that Code Case N-580-2 revises the heat treatment temperature range of Alloy 600 with columbium added from a range of 1,950 to 2,000 °F to a range of 1,875 to 2,000 °F. This revision is based on test data indicating that the material properties and grain size of Alloy 600 heat-treated at 1,900 °F are superior to those properties at higher heat treatment temperatures. The staff reviewed Code Case N-580-2 and found it acceptable because it improves the heat treating requirements currently listed in N-580-1. This change from SB-166 and SB-167 with modified niobium to niobium-modified Alloy 600, in accordance with Code Case N-580-2, does not modify the materials previously approved in the ABWR DCD. Instead, the change adds additional requirements that are acceptable to the staff. The staff therefore found the applicant's use of Code Case N-580-2 acceptable.

With regard to the applicant's use of superseded Code Cases N-71-15 and N-319, the applicant states that Code Cases N-71-15 and N-319 are listed in ABWR DCD Tier 2, Table 5.2-1, as applicable to equipment in systems within the scope of the DCD. STP FSAR Tier 2 Table 5.2-1 lists Code Cases N-71-18 and N-319-3 for applicability to equipment in systems outside the scope of the ABWR DCD (i.e., site-specific systems). As part of STD DEP 1.8-1, the applicant has modified Section 1.8 of the ABWR DCD to state that the STP Units 3 and 4 FSAR conforms to RG 1.84, Revision 33. The staff notes that requirements related to the implementation of the Code cases are in 10 CFR 50.55a(b). RG 1.84 states that when a licensee initially applies a Code case listed in Table 1, "Acceptable Section III Code Cases," or Table 2 "Conditionally Acceptable Code Cases," the licensee must implement the most recent version of that Code case incorporated by reference in 10 CFR 50.55a. The staff therefore found the applicant's use of superseded Code cases unacceptable. The staff issued **RAI 05.02.01.02-5** requesting the applicant to delete superseded Code cases from Table 5.2-1 and to list the most recent revisions of Code cases listed in RG 1.84, Revision 33.

The applicant's response to RAI 05.02.01.02-5 dated January 18, 2010 (ML100191523), states that the superseded Code Cases N-71-15 and N-319 listed in Table 5.2-1 will be deleted in the next revision of the FSAR. The staff found this response acceptable, and tracked this issue as **Confirmatory Item 05.02.01.02-5**. The applicant subsequently submitted COL FSAR Revision 4, which the staff reviewed. The staff verified that Code Cases N-71-15 and N-319 have been deleted from FSAR Table 5.2-1. RAI 05.02.01.02-5 and its associated confirmatory item are therefore resolved.

The staff reviewed the STP COL application and determined that STP FSAR Subsection 5.2.1.2 appropriately incorporates by reference ABWR DCD Tier 2 Subsection 5.2.1.2, in satisfying the NRC regulations for the design, fabrication, erection, testing, and inspection of plant SSCs commensurate with the importance of the safety function to be performed, by referencing the use of the accepted ASME Code cases. As a result, the staff found that compliance with the provisions of the ASME Code cases accepted in RGs 1.84, 1.147, and 1.192 (or individually reviewed and accepted by the NRC staff) will result in the quality of the component that is commensurate with the importance of the safety functions of the components at STP Units 3 and 4 that satisfies the requirements of GDC 1.

5.2.1.2.5 Post Combined License Activities

There are no post COL activities related to this subsection.

5.2.1.2.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information relating to the applicable ASME Code cases. No outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues related to ASME Code Cases that were incorporated by reference have been resolved.

The staff confirmed that STP FSAR Subsection 5.2.1.2 adequately incorporates by reference ABWR DCD Tier 2 Subsection 5.2.1.2. The staff's review confirmed that the applicant has adequately addressed the Tier 2* departure in accordance with 10 CFR Part 50, Appendix A, GDC 1, 10 CFR 50.55a, and Subsection 5.2.1.2 of NUREG–0800.

5.2.2 Overpressure Protection

5.2.2.1 *Introduction*

Section 5.2.2, "Overpressure Protection," of the FSAR addresses the safety and relief valves (SRVs) and the portion of the reactor protection system that ensures overpressure protection for the RCPB during operation at power.

5.2.2.2 *Summary of Application*

Section 5.2.2 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.2.2 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in COL FSAR Section 5.2.2, the applicant provides the following:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP Vendor

This departure describes the applicant's decision to use the services of an alternate vendor to support the application.

5.2.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the overpressure protection, and the associated acceptance criteria, are in Section 5.2.2 of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 2 departures. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

5.2.2.4 *Technical Evaluation*

As documented in NUREG–1503, the staff reviewed and approved Section 5.2.2 of the certified ABWR DCD. The staff reviewed Section 5.2.2 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[4] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the overpressure protection.

The staff reviewed the information in the COL FSAR:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP Vendor

This departure replaces terms such as General Electric (GE) and GE-Hitachi (GEH) with the generic term nuclear steam supply system (NSSS) vendor, or a specified alternative vendor, or eliminates these terms altogether in some cases.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval.

5.2.2.5 *Post Combined License Activities*

There are no post COL activities related to this subsection.

5.2.2.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the overpressure protection that were incorporated by reference have been resolved.

The staff found it reasonable that the identified Tier 2 departures are characterized as not requiring prior NRC approval per 10 CFR Part 52, Appendix A, Section VIII.B.5.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 *Introduction*

This FSER section addresses information related to the materials selection, fabrication, and processing of RCPB piping and components, as well as the compatibility of RCPB materials with the reactor coolant.

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[4] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

5.2.3.2 *Summary of Application*

Section 5.2.3 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.2.3 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 5.2.3, the applicant provides the following:

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP Admin

In COL FSAR Subsection 5.2.3.2.2.3, the applicant modifies as an STD DEP Admin the information supplied in the corresponding ABWR DCD subsection. The ABWR DCD requires condenser tubes and tube sheets to be made of titanium alloys. The applicant modifies this information to state that condenser tubes and tube sheet materials are specified in Subsection 10.4.1.2.3.

- STD DEP 4.5-1 Reactor Materials

In COL FSAR Subsection 5.2.3.2.3, the applicant lists STD DEP 4.5-1. As part of the aforementioned departure, the applicant modifies ABWR DCD Subsections 5.2.3.2.3(1) and (2) to add and to delete from the list of stainless steels materials that are exposed to the reactor coolant. The applicant modifies ABWR DCD Table 5.2-4, "Reactor Coolant Pressure Boundary Materials," as part of STD DEP 4.5-1. Modifications include editorial changes to provide clarity, to correct typographical errors, and to identify some different materials used in the RCPB.

5.2.3.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503. In addition, the relevant requirements of the Commission regulations for RCPB materials, and the associated acceptance criteria, are in Section 5.2.3 of NUREG–0800.

The applicable regulatory requirements for materials specifications are in 10 CFR 50.55a, which endorses ASME Code Section III. These requirements for materials specifications are met by complying with the applicable provisions of the ASME Code and by the acceptable application of materials Code cases, as described in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III." Meeting the above specifications fulfills the requirements of 10 CFR Part 50, Appendix A, GDC 1 and GDC 30.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 2 departures. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

5.2.3.4 *Technical Evaluation*

As documented in NUREG–1503, the NRC staff reviewed and approved Section 5.2.3 of the certified ABWR DCD. The staff reviewed Section 5.2.3 of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[5] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the RCPB materials.

The staff reviewed the information in the COL FSAR:

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP Admin

ABWR DCD Tier 2 Subsection 5.2.3.2.2.3, "Source of Impurities," requires condenser tubes and tube sheets to be made of titanium alloys. ABWR DCD Tier 2 Subsection 10.4.1.2.3 states, "The condenser and water boxes are all welded carbon steel or low alloy ferritic steel. The tubes are stainless steel or titanium with compatible stainless steel or titanium carbon steel clad tube sheets depending on circulating water quality." Due to the apparent inconsistencies between ABWR DCD Subsections 5.2.3.2.2.3 and 10.4.1.2.3, the applicant modified COL FSAR Subsection 5.2.3.2.2.3 to state that "condenser tubes and tube sheet materials are specified in Subsection 10.4.1.2.3."

The applicant lists this departure as STD DEP ADMIN and states in Part 07, "Departures Report," of the application that this departure meets the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5 and does not require prior NRC approval. NRC staff determined that the departure adequately corrects inconsistencies between the ABWR DCD subsections referenced above and meets the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval. In addition, the applicant's process for evaluating departures and changes to the DCD are subject to NRC inspections.

- STD DEP 4.5-1 Reactor Materials

As part of STD DEP 4.5-1, the applicant proposed to provide additional information in ABWR DCD Subsection 5.2.3.2.2.3, "Source of Impurities"; Subsection 5.2.3.2.3, "Compatibility of Construction Materials with Reactor Coolant"; and Table 5.2-4, "Reactor Coolant Pressure Boundary Materials." Most of the modifications to Table 5.2-4 clarify product form, material type, and component definition and do not involve changes in the material types that will be used. These modifications are described appropriately in the applicant's proposed revision (in response to RAI 05.02.03-1) to the description of STD DEP 4.5-1. However, the applicant's Departures Report summary does not appropriately address these proposed changes. In addition, the proposed modifications to Table 5.2-4 that are discussed in the Departures Report reference equivalent materials. It is the staff's understanding that equivalent materials apply only to non-ASME Code components. In addition, changes in materials made to Table 5.2.4 are not described in the applicant's Departures Report. Therefore, the staff issued **RAI 05.02.03-2** requesting the applicant to modify the Departures Report, as it relates to STD DEP 4.5-1, to address all departures from ABWR DCD Sections 4.5.1, 4.5.2, and 5.2.3 and Table 5.2-4. The staff also requested the applicant to provide clarity by dividing the Departures Report into separate sections for each section referenced above. The applicant's response to this RAI dated December 30, 2009 (ML100050182), provides a description and evaluation summary of

[5] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

each modification to ABWR DCD Sections 4.5.1, 4.5.2, and 5.2.3 and Table 5.2-4 as identified in STD DEP 4.5-1. The applicant's response also states the intent to modify the Departures Report as requested by the staff. The staff tracked the above issue as **Open Item 05.02.03-2** in the SER with open items. The applicant provided the proposed modifications to the Departures Report in a letter dated January 25, 2010 (ML100290011). The staff reviewed the applicant's proposed modifications and found them acceptable, because the applicant will list a description of the changes and an evaluation summary of each change made to ABWR DCD Sections 4.5.1, 4.5.2, and 5.2.3 and Table 5.2-4 as part of STD DEP 4.5-1. In addition, the proposed change to the Departures Report clarifies that equivalent materials are not used in RCPB materials listed in Table 5.2-4. The staff verified that the appropriate modifications described in the applicant's response to RAI 05.02.03-2 were made in COL FSAR Revision 4, Part 7 Section 3.0, STD DEP 4.5-1. Based on the above information, the staff found the applicant's response acceptable. RAI **05.02.03-2** is therefore resolved.

5.2.3.5 *Post Combined License Activities*

There are no post COL activities related to this section.

5.2.3.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to the RCPB materials, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues related to the RCPB materials that were incorporated by reference have been resolved.

The staff found it reasonable that the identified Tier 2 departure is adequately characterized as not needing prior NRC approval per 10 CFR Part 52, Appendix A, Section VIII.B.5.

5.2.4 Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary (Related to RG 1.206, Section 5.2.4, "Inservice Inspection and Testing of Reactor Coolant Pressure Boundary")

5.2.4.1 *Introduction*

This section of the FSAR addresses RCPB components that must be designed to permit periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity. This section will also assess the COL applicant's PSI/ISI Program for Class 1 components, as well as the PSI/ISI Operational Program, because the staff approved the PSI/ISI Program for only the reactor pressure vessel (RPV) at the design certification stage.

5.2.4.2 *Summary of Application*

Section 5.2.4 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.2.4 of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in COL FSAR Section 5.2.4, the applicant provides the following:

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP Vendor

This departure addresses the transfer of the design certification responsibility to Toshiba for the design of the RPV. The remainder of the PSI/ISI Program is the responsibility of the COL applicant.

- STD DEP 5.2-2 PSI/ISI NDE of the Reactor Coolant Pressure Boundary

This departure addresses the addition of the straight length of piping for accessibility to perform nondestructive examinations (NDEs), PSI of Class 1 components and welds, and components subject to exclusion from ASME Section XI.

- STD DEP 5A-1 Delete Appendix on Compliance with Regulatory Guide 1.150

This departure addresses the ultrasonic testing (UT) qualification requirements for UT systems used to examine the RPV. This departure also addresses the general conduct of UT examinations.

COL License Information Item

- COL License Information Item 5.2 Plant Specific ISI/PSI

This COL license information item provides supplemental information to address the plant-specific PSI/ISI Program attributes as required by ASME Code Edition and Addenda, qualification and conduct of UT systems, submittal of relief requests, and operational aspects of the PSI/ISI Program.

5.2.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the PSI/ISI testing of the RCPB, and the associated acceptance criteria, are in Section 5.2.4 of NUREG-0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 2 departures. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

The regulatory bases for acceptance of the resolution to the departures and supplementary information on ISI of the Class 1 coolant boundary are established in 10 CFR 50.55a, as it pertains to specification of the preservice and periodic inspection and testing requirements of the ASME Code for Class 1 systems and components.

5.2.4.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 5.2.4 of the certified ABWR DCD. The staff reviewed Section 5.2.4 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[6] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

The staff's review of this application includes departures, COL License Information Item 5.2, and the PSI/ISI Operational Program for Class 1, 2, and 3 components, as summarized below.

<u>Tier 2 Departure Not Requiring Prior NRC Approval</u>

- STD DEP Vendor

The COL applicant states that Toshiba is responsible for designing the RPV for accessibility to perform PSIs and ISIs. Responsibility for designing other components for these inspections belongs to the COL applicant, along with the responsibility for specifying the Edition of ASME Section XI to be used based on the procurement date of the component, per 10 CFR 50.55a. The COL applicant also states that the ASME Code requirements discussed in this section are based on the Edition of ASME Section XI specified in Table 1.8-21. However, the COL applicant does not include a discussion of Table 1.8-21a in this departure, which specifies the 2004 Edition of ASME Section XI for site-specific components. If the design of the RPV complies with the 1989 Edition while the PSI/ISI Program complies with the 2004 Edition of ASME Section XI, the COL applicant should make the difference clear in the departure. NRC staff was unable to determine the acceptability of the applicant's evaluation of this departure per the requirements in 10 CFR Part 52, Appendix A, Section VIII.B.5. The staff thus issued **RAI 05.02.04-3** requesting additional information from the applicant.

The applicant's response to RAI 05.02.04-3 dated July 23, 2009 (ML092080080), states that STP Units 3 and 4 will be fully compliant with the requirements of 10 CFR 50.55a with regard to PSI and ISI examinations. The PSI/ISI Program will be based on the editions and addenda of Section XI of the ASME Code incorporated by reference in paragraph (b), on the date 12 months before the date scheduled for initial loading of fuel, under the combined license per 10 CFR Part 52. The applicant adds that FSAR Subsection 5.2.6.2 will be revised to state that the PSI/ISI Program is based on the 2004 Edition of the ASME Code. The staff concluded that the changes proposed by the applicant are in agreement with the regulations and are therefore acceptable. This RAI was tracked as **Confirmatory Item 05.02.04-3**. The staff confirmed in revision 4 of the FSAR that the proposed changes were included in the COL application. Therefore RAI 05.02.04-3 and Confirmatory Item 05.02.04-3 are closed.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that the departure does not require prior NRC

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[6] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

approval. The applicant's process for evaluating departures from the certified ABWR DCD is subject to NRC inspections.

- STD DEP 5.2-2 PSI/ISI NDE of the Reactor Coolant Pressure Boundary

The COL applicant states that straight lengths of pipe and spool pieces shall be added between fittings with the minimum length determined by the formula L = 2T + 152 mm, where L equals the length of the spool piece and T equals the pipe wall thickness. The COL applicant also states that in instances where less than the minimum straight length is used, “an evaluation is performed to ensure that sufficient access exists to perform the required examinations.”

The requirements in 10 CFR 50.55a(g)(3) state (in part) that Class 1, 2, and 3 components and supports should be designed and provided with inservice examinations. Because sufficient access is incorporated into the design to perform the required examinations, the staff concluded that this portion of the subject Departure is in compliance with the regulation and ASME Section XI and is therefore acceptable.

The COL applicant states that PSI examinations will be performed on 100 percent of the Class 1 pressure-retaining welds in accordance with IWB-2200, with the exception of welds excluded by ASME Section XI such as ASME lines smaller than nominal pipe size (NPS) 1, volumetric examination of lines smaller than NPS 4, visual VT-3 examination of valve body and pump casing internal surfaces, and the visual VT-2 examination for Code Category BP - All Pressure Retaining Components and Code Category BE - Pressure Retaining Partial Penetration Welds in Vessels. The COL applicant also states that if the design incorporates external category B-O CRD (CRD) housing welds, the PSI examination will be extended to include 100 percent of the welds in the installed peripheral CRD housings, in accordance with IWB-2200.

The staff reviewed the proposed changes with ASME Section XI, 2004 Edition. The staff found that the changes are in compliance with the ASME requirements for exclusions, extent, and type of examinations. Because the proposed changes are in compliance with ASME Section XI, the staff concluded that this portion of the departure is therefore acceptable.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that the departure does not require prior NRC approval. The applicant's process for evaluating departures from the certified ABWR DCD is subject to NRC inspections.

- STD DEP 5A-1 Delete Appendix on Compliance with Regulatory Guide 1.150

The COL applicant deletes the portions of the ABWR DCD referring to the use of the GE reactor vessel inspection system, which meets the guidance of RG 1.150. The applicant states that the UT system for examination of the reactor vessel meets the qualification requirements in Subsection 5.2.4.3.4. ABWR DCD Subsection 5.2.4.3.4 states that the personnel and equipment used for UT examination shall be qualified in accordance with ASME Section XI, Appendix VII and Appendix VIII.

The requirements in 10 CFR 50.55a state that personnel, equipment, and procedures should be qualified in accordance with ASME Section XI, Appendix VII and VIII. Furthermore, COL

License Information Item 5.2 states that NRC requirements demonstrating performance addressed by RG1.150 will be conducted in accordance with ASME Section XI, Appendix VIII, as required by 10 CFR 50.55a. NRC staff concluded that the changes are consistent with the change in the vendor supplying the RPV addressed under Departure STD DEP "Vendor," and the testing requirements are in compliance with the regulations under 10 CFR 50.55a, which replace RG 1.150 with Appendix VII and VIII. The subject departure is in compliance with the regulations and is therefore acceptable.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that the departure does not require prior NRC approval. The applicant's process for evaluating departures from the certified ABWR DCD is subject to NRC inspections.

COL License Information Item

- COL License Information Item 5.2 Plant Specific ISI/PSI

The COL applicant states that the PSI/ISI Program will be based on the 1989 Edition of ASME Section XI, as identified in Table 1.8-21. This item is discussed under RAI 05.02.04-3. The applicant also states that UT examinations of the RPV will be qualified in accordance with ASME Section XI, Appendix VIII, which is required under 10 CFR 50.55a. The qualification of UT examinations in accordance with Appendix VIII is discussed above under Departure STD DEP 5A-1, which the staff found acceptable. This RAI was tracked as **Confirmatory Item 05.02.04-3** to confirm that the proposed revision was included in an upcoming revision of the COL FSAR. The staff confirmed in Revision 4 of the FSAR that the proposed changes were included in the COL application. Therefore RAI 05.02.04-3 and Confirmatory Item 05.02.04-3 are closed.

The COL applicant states that the conduct of UT examinations will be in accordance with Appendices I and VIII, which address the near-surface examination and surface resolution, including the use of electronic gating as well as internal surface examination. Appendix I is a mandatory appendix required under ASME Section XI. The staff concluded that the proposed change is in compliance with ASME Section XI and is therefore acceptable.

The COL applicant states that Code Cases are listed in Table 5.2-1. The staff compared the Code Cases with those approved by the staff under RG 1.147 and found that the Code Cases are approved for use. The regulations in 10 CFR 50.55a allow for the use of Code Cases for a PSI/ISI Program, provided that all related requirements are met. The staff concluded that this portion of the COL license information item is acceptable because it is in compliance with the regulation.

The COL applicant states that any additional relief requests shall be submitted with a supporting technical justification, if needed.

ABWR DCD Subsection 5.2.4.2.2 states that the physical arrangement of piping, pumps, and valves provides personnel access to each weld location for the performance of UT and surface examinations and sufficient access to supports for the performance of visual VT-3 examinations. In addition, 10 CFR 50.55a(g)(3) requires Class 1, 2 , and 3 components to be designed to enable the performance of ISI examinations. The staff expected interferences from design, geometry, and materials of construction to be eliminated at the design stage so that no relief requests are necessary for the first 120-month ISI interval. The discussion of additional relief requests parenthetically implies that the regulation may not be met. Based on this concern, the staff issued **RAI 05.02.04-2**.

The applicant's response to RAI 05.02.04-2 dated July 23, 2009 (ML092080080), states that STP Units 3 and 4 will be fully compliant with the requirements of 10 CFR 50.55a with regard to PSI and ISI examinations. The reference "to the extent practical" appears in 10 CFR 50.55a(g)(4) and applies to 10 CFR 50.55a(g)(4)(i), which relates to ISI requirements for the initial 120-month ISI interval. The applicant adds that the Code requirements for (g)(4) may be different from the construction ASME Code of Record (2004). Although the applicant does not expect any relief requests to be necessary, the applicant states that it would be impractical to commit to no relief requests based on a Code edition that was issued subsequent to applying it to the component design. The staff agreed with the COL applicant's interpretation of the regulations and concluded that the design will incorporate the necessary attributes to eliminate interferences to the performance of ISI examinations due to design, geometry, and materials of construction. The staff agreed that any changes to the examination requirements for the initial inspection interval will be minimal, with few requests for relief from ISI requirements. The COL applicant's response is in compliance with the regulations and is therefore acceptable.

The COL applicant states that the PSI/ISI Program for the RCPB is described in Section 5.2.4 and Table 5.2-8. The staff will evaluate the PSI/ISI Operational Program elements of ABWR DCD Section 5.2.4 and Table 5.2-8 later in this FSER.

The COL applicant states that the commitment to provide a comprehensive site-specific PSI/ISI Program plan to the NRC at least 12 months before respective unit commercial power operation is discussed in Section 6.6.9.1. The regulation in 10 CFR 50.55a(g)(4)(i) states that ISI examinations and pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code (or Code cases) incorporated by reference in paragraph (b) of this section on the date 12 months before the date scheduled for initial loading of fuel, under a COL under Part 52 of this chapter and subject to the limitations and modifications listed in paragraph (b) of this section. The proposed change is in compliance with the regulation. Therefore, the staff considered this portion of the COL license information item acceptable, and it is discussed in further detail under Section 6.6 of this SER.

PSI/ISI Operational Program

Operational programs are specific programs required by regulations. The COL application should fully describe operational programs as defined in SECY-05-0197. In addition, COL applicants should provide schedules for implementing milestones of these operational programs. The PSI/ISI Programs are identified as operational programs in RG 1.206. This section of the FSER addresses the PSI/ISI Operational Programs for ASME Code Class 1, 2, and 3 components that the COL applicant incorporates by reference from the ABWR DCD for Class 1, 2, and 3 components.

For STP Units 3 and 4 COL FSAR Revision 2, the applicant incorporates by reference the PSI/ISI Program descriptions from ABWR DCD, Sections 5.2.4 and 6.6. As discussed in RG 1.206, a fully described PSI/ISI Program should address (1) the system boundary subject to inspection, (2) accessibility, (3) examination categories and methods, (4) inspection intervals, (5) evaluation of examination results, (6) system pressure tests, (7) Code exemptions, (8) relief requests, and (9) ASME Code cases.

The ABWR DCD states that the ASME Code requirements discussed in this section are provided for information and are based on the Edition of ASME Section XI specified in Table 1.8-21. STP Units 3 and 4 COL FSAR Revision 2, Table 1.8-21a specifies the 2004 Edition of ASME Section XI to determine the requirements for the initial interval in the ISI Program. The PSI/ISI shall meet the requirements set forth in Section XI of the ASME Code, as specified in 10 CFR 50.55a(b)(2) with limitations and modifications therein.

System Boundary Subject to Inspection

The SRP states that the applicant's definition of the system boundary is acceptable if for a boiling-water reactor (BWR), the inspection requirements of 10 CFR 50.55a are met for all Class 1, 2, and 3 components (and their supports). The system boundary as defined in 10 CFR 50.2 includes all pressure vessels, piping, pumps, and valves connected to the reactor coolant system, up to and including the outermost containment isolation valve in system piping that penetrates the primary reactor containment. This is the second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment and the reactor coolant system safety and relief valves. NRC staff reviewed the system boundary description under Sections 5.2.4 and 6.6 for Class 1, 2, and 3 components. The staff found that the descriptions for the component boundaries and exclusions are in compliance with the requirements under ASME Section XI and 10 CFR 50.55a. The staff concluded that the inspection requirements will be met for the appropriate Class 1, 2, and 3 components and are therefore acceptable.

Accessibility of Systems and Components

The SRP states that the design and arrangement of system components are acceptable if there is adequate clearance in accordance with ASME Code Section XI, Subarticle IWA-1500, "Accessibility." In addition, 10 CFR 50.55a(g)(3) requires Class 1, 2, and 3 components and supports to be designed with access to ISI of these components, which must meet the PSI requirements set forth in the Editions and Addenda of Section XI for the ASME Code of Record.

ABWR DCD Tier 2, Subsection 5.2.4.1.1 describes accessibility for inspection and states that the physical arrangement of ASME Code Class 1 components is designed to allow personnel and equipment access to perform the required ISIs specified by ASME Section XI, Table IWB-2500. The piping arrangement allows for an adequate separation of piping welds so that space is available to perform the ISI by incorporating the equation $L = 2T + 152$ mm, where L equals the length of spool piece and T equals the pipe wall thickness to determine the amount of straight sections of pipe and spool pieces added between fittings. The applicant states that welds are located to permit UT examination from at least one side, but where component geometries permit, there is access from both sides. ABWR DCD Section 6.6.2 states that the COL applicant is responsible for the PSI/ISI design for inspectability, and that the PSI will be completed during fabrication for Class 2 residual heat removal (RHR) heat exchangers. The applicant states that during the ISI examination, the heat exchanger nozzle-to-shell welds may have limited access, thus requiring the submittal of a relief request to the staff. In the area of one-sided access, the staff was not able to determine whether the UT procedures used on a single-sided ferritic vessel and piping and stainless steel piping would meet the qualification requirements of 10 CFR 50.55a(b)(2)(xvi) and whether 10 CFR 50.55a(g)(3) is incorporated into the PSI/ISI Operational Program that the COL applicant is responsible for. Based on these concerns, the staff issued **RAI 05.02.04-4**.

In the response to RAI 05.02.04-4 dated September 24, 2009 (ML092710233), the applicant states that the PSI/ISI Operational Program will be in compliance with 10 CFR 50.55a(b)(2)(xvi) and 10 CFR 50.55a(g)(3). The staff found that the applicant's response is in agreement with the regulations and is therefore acceptable.

Examination Categories and Methods

The SRP states that the examination categories and methods specified in the ABWR DCD are acceptable if they meet the requirements in ASME Code Section XI, Articles IWB-2000, IWC-, and IWD-2000, "Examination and Inspection." Every area subject to examination falling within one or more of the examination categories in Article IWB-2000 must be examined at least to the extent specified. The requirements of Article IWB-2000 also list the methods of examination for the components and parts of the pressure-retaining boundary.

Furthermore, the applicant's examination techniques and procedures used for the PSI/ISI of the system are acceptable if they meet the following criteria:

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

- The methods, procedures, and requirements regarding qualifications of nondestructive examination personnel are in accordance with Article IWA-2300, "Qualification of Nondestructive Examination Personnel."

- The methods, procedures, and requirements regarding qualifications of personnel performing UT 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 UT examination systems reflects the requirements in Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Division 1 of ASME Code Section XI.

ABWR DCD Tier 2 Subsection 5.2.4.3 and Section 6.6.3, "Examination Categories and Methods," discuss the examination techniques, categories, and methods. NRC staff found that the ABWR DCD lists the appropriate subarticles of IWA-2000 for conducting visual, surface, and volumetric examinations, thus meeting the SRP acceptance criteria. In addition, the ABWR DCD lists examples of PSI/ISI Program visual, surface, and volumetric examination techniques, boundaries, and components in Tables 5.2-8 and 6.6-1 for Class 1, 2, and 3 components. The staff compared the subject tables to Tables IWB-, IWC-, and IWD-2500 of ASME Section XI and found that the ABWR DCD is in compliance with ASME Section XI. Furthermore, the ABWR DCD states that personnel performing examinations shall be qualified in accordance with ASME Section XI Appendix VII, with the systems qualified to Appendix VIII. Table 1.8-21a, indicates that the baseline Code used for the ABWR DCD is the 2004 Edition of ASME Section XI, which requires the implementation of mandatory Appendices VII for the qualifications of NDE personnel for UT examination, and VIII for demonstrating the performance of the UT examination for the reactor pressure boundary piping, reactor vessel (RV) welds, and RV head bolts. Furthermore, Appendix VII modifies IWA-2300 for the qualifications of NDE personnel. The staff concluded that the ABWR DCD examples are in compliance with the requirements of ASME Section XI and the SRP acceptance criteria and are therefore acceptable.

Inspection Intervals

The required examinations and pressure tests must be completed during each 10-year interval of service, hereinafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of Article IWA-2000, "Examination and Inspection," concerning inspection intervals of ASME Code Section XI. ABWR DCD Tier 2 Subsection 5.2.4.4 and Section 6.6.4, "Inspection Intervals," discuss inspection intervals, which are defined in Subarticles IWA-2400 and IWB-2400 of ASME Code Section XI. The inspection intervals specified for the PSI/ISI Operational Program are consistent with the definitions in Section XI of the ASME Code and are therefore acceptable.

Evaluation of Examination Results

The SRP states that "standards for evaluation of examination results are acceptable if they are in accordance with the requirements of ASME Code Section XI, Article IWB-3000, 'Acceptance Standards'." The SRP also states that the proposed program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications is acceptable if the program agrees with the requirements of ASME Code Section XI, Article IWA-4000, "Repair/Replacement Activities." The criteria that establish the need for repair or replacement are described in ASME Code Section XI, Article IWB-3000, "Acceptance Standards."

ABWR DCD Tier 2 Subsection 5.2.4.5 and Section 6.6.5, "Evaluation of Examination Results," discuss the evaluation of examination results. Examination results are evaluated according to ASME Code, Section XI, IWB-3000, with flaw indications evaluated according to IWB-3400 and Table IWB-3410-1. Repair procedures, if required, are evaluated according to ASME Code Section XI, IWB-4000. NRC staff found that the corresponding ASME Section XI evaluation criteria are noted in the ABWR DCD for Class 2 and 3 components. Based on this method of evaluating examination results, and the use of the appropriate ASME Code rules for repair, the applicant's evaluation of examination results for Class 1, 2, and 3 components meets the SRP acceptance criteria and is therefore acceptable.

System Pressure Tests

The pressure-retaining Code Class 1 Component Leakage and Hydrostatic Pressure Test Program is acceptable if the program is in accordance with the requirements of Section XI Article IWB-5000 and the technical specification requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. The pressure tests verify pressure boundary integrity in conjunction with the ISI.

ABWR DCD Tier 2 Subsection 5.2.4.6, "System Leakage and Hydrostatic Pressure Tests," and Section 6.6.6, "System Pressure Tests," state that Class 1, 2, and 3 systems and components are pressure tested in accordance with IWB-5000, IWC-5221, and IWD-5221, respectively, of the ASME Code, at the maximum operating temperature and pressure indicated in the applicable process flow diagram for the system. Because the applicant's methodology for performing pressure testing of the Class 1, 2, and 3 boundary and components is in compliance with the ASME Code and agrees with the SRP acceptance criteria, the methodology for performing system pressure testing is therefore acceptable.

Code Exemptions

The SRP states that exemptions from Code examinations should be permitted if the criteria in Subarticles IWB-, IWC-, and IWD-1220, "Components Exempt from Examination," are met.

ABWR DCD Tier 2 Subsection 5.2.4.1.6 and Section 6.6.8 state that ASME Section XI Code exemptions are permitted by Subarticles IWB-, IWC-, and IWD-1220. Furthermore, the specified ABWR DCD allowable ASME Code exemptions are based on the design in Tables 5.2-8 and 6.6-1 for ASME Class 1, 2, and 3 components. NRC staff found that the ASME Code exemptions listed in the DCD are in compliance with ASME Section XI and are therefore acceptable.

Relief Requests

The SRP states that the reviewer will determine whether the licensee has demonstrated that any ASME Code requirement is impractical due to design, geometry, or materials of construction. NRC staff found that no part of the ABWR DCD or STP Units 3 and 4 COL FSAR discusses the use of relief requests as part of the PSI/ISI Operational Program. Based on this concern, the staff issued **RAI 05.02.04-1**.

The applicant's response to RAI 05.02.04-1 dated July 23, 2009 (ML092080080), states that STP Units 3 and 4 will be fully compliant with the requirements of 10 CFR 50.55a with regard to PSI and ISI examinations. The PSI/ISI Program will be based on the editions and addenda of Section XI of the ASME Code incorporated by reference in paragraph (b) on the date 12 months before the date scheduled for initial loading of fuel under the combined license under 10 CFR Part 52. The applicant adds that FSAR Subsection 5.2.6.2 will be revised to state that the PSI/ISI Program is based on the 2004 Edition of the ASME Code. The staff concluded that the changes proposed by the applicant are in agreement with the regulations and are therefore acceptable. This RAI was tracked as **Confirmatory Item 05.02.04-1**. The staff confirmed in revision 4 of the FSAR that the proposed changes were included in the COL application. Therefore RAI 05.02.04-1 and Confirmatory Item 05.02.04-1 are closed.

Code Cases

The SRP acceptance criteria states that ASME Code Cases are reviewed for acceptability and compliance with RG 1.147. NRC staff reviewed STP Units 3 and 4 COL FSAR Table 5.2-1, "Reactor Coolant Pressure Boundary Components Applicable Code Cases," and found that the Code Cases listed comply with RG 1.147 and are therefore acceptable.

Augmented ISI to Protect Against Postulated Piping Failures

The SRP states that the ISI Program is reviewed to verify that the high energy system piping between containment isolation valves receives an augmented ISI that meets four criteria: (1) weld accessibility, (2) 100 percent volumetric examination of circumferential and longitudinal welds within the boundary, (3) inspection ports installed in guard pipes, and (4) areas subject to examination will be in accordance with IWC-2000.

ABWR DCD Subsection 6.6.7.1 states that 100 percent of circumferential welds will be ultrasonically examined each interval, and no longitudinal welds are present in the seamless piping. In addition, there is accessibility for all the subject welds that are inspected in accordance with IWC-2000 and the nondestructive methodologies listed in the PSI/ISI Program.

NRC staff concluded that the accessibility requirement negates the need for inspection ports, and the augmented Inspection Program meets the SRP acceptance criteria and is therefore acceptable.

Erosion-Corrosion Program

The erosion-corrosion is evaluated under Section 6.6 of this SER.

Combined License Information Items

At the COL application stage the PSI/ISI Programs were not developed, but they will in fact be developed during the construction phase. RG 1.206, Section C.III.1, Chapter C.I.5.2.4.1, for the RCPB applies to the PSI/ISI Program, which states that the detailed procedures for performing the examinations may not be available at the time of the COL application, and the COL applicant should make a commitment to provide sufficient information to demonstrate that the procedures meet ASME Code standards. This information should be provided at a predetermined time agreed upon by both parties. In order for the staff to obtain a reasonable assurance finding of the acceptability of the PSI/ISI Operational Programs, the staff must be able to inspect the construction of the plant for conformance to the regulations and to the ASME Code of Record. The staff is currently drafting a generic license condition for all COL applicants to submit a schedule that enables the staff to inspect the PSI/ISI Program during the construction phase. Based on the acceptance of the license condition, the COL License Information Item 5.2 (shown below) is acceptable to the staff.

Combined License Information Items

Item No.	Description	Section	Action Required By COL Applicant	Action Required By COL Holder
13.4(1)	A COL applicant referencing ABWR design certification will fully describe the operational programs, as defined in SECY-05-0197, and will provide commitments for the implementation of operational programs required by the regulation. In some instances, programs may be implemented in phases, which the COL applicant is to include in submittals.	5.2.4	Y	

5.2.4.5 *Post Combined License Activities*

There no post COL activities related to this section.

5.2.4.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to the PSI/ISI and testing of RCPB. No outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the ISI and testing of RCPB that were incorporated by reference have been resolved.

The staff found it reasonable that the identified Tier 2 departures are characterized as not requiring prior NRC approval per 10 CFR Part 52, Appendix A, Section VIII.B.5.

The staff's review confirmed that the applicant has adequately addressed the COL license information in accordance with 10 CFR 50.55a and Section 5.2.4 of NUREG–0800.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

5.2.5.1 *Introduction*

This section of the FSAR, addresses RCPB leakage detection and isolation. The leakage detection system (LDS) consists of temperature, pressure, radiation and flow sensors with associated instrumentation, power supplies, and logic used to detect, indicate, and alarm leakage from the reactor primary pressure boundary and, in certain cases, to initiate closure of isolation valves to shut off leakage external to the containment. Abnormal leakage from systems within the primary containment (drywell) and within selected areas of the plant outside the drywell (both inside and outside the reactor building) is detected, indicated, alarmed, and, in certain cases, isolated. The system is designed to conform to RG 1.45, "Reactor Coolant

Pressure Boundary Leakage Detection Systems" (for leak detection functions), and the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std)-279, "Criteria for Protection Systems for Nuclear Power Generating Stations," (for isolation function).

5.2.5.2 *Summary of Application*

Section 5.2.5 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.2.5 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in COL FSAR Section 5.2.5, the applicant provides the following:

<u>*Tier 1 Departures*</u>

- STD DEP T1 2.3-1 — Deletion of MSIV Closure and Scram on High Radiation

This departure describes the elimination of the main steam isolation valve (MSIV) automatic closure and reactor scram on the high main steam line radiation monitor (MSLRM) indication. The applicant has determined that this departure requires NRC review and approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.A.4.

- STD DEP T1 2.4-2 — Feedwater Line Break Mitigation

This departure describes the addition of logic to cause a trip of the condensate pumps upon an indication that a feedwater line break (FWLB) in the drywell has occurred. The applicant has determined that this departure requires NRC review and approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.A.4.

<u>*Tier 2 Departure Requiring Prior NRC Approval*</u>

- STD DEP 7.3-12 — Leak Detection and Isolation System Sump Monitoring

This departure describes increasing the total leakage (averaged over a 24-hour period) from 95 L/min to 114 L/min; increasing the unidentified leakage from 3.785 L/min to 19 L/min; and adding a limit for unidentified leakage increase of 8 L/min within the previous 4-hour period in Mode 1. This departure affects Technical Specifications (TS). The applicant has determined that this departure requires NRC review and approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.C.4.

<u>*Tier 2 Departure Not Requiring Prior NRC Approval*</u>

- STD DEP 7.3-11 — Leak Detection and Isolation System Valve Leakage Monitoring

This departure describes the elimination of the piping and instrumentation for leakage detection from valve stems of large-bore reactor coolant pressure boundary isolation valves. The applicant has classified this departure as one not requiring NRC review and approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5.

In addition, in FSAR 5.2.6, "COL License Information," the applicant provides the following:

COL License Information Items

- COL License Information Item 5.1 Conversion of Indicators

This COL license information item addresses the requirement for the applicant to provide procedures and graphs for operations to convert the various indicators into a common leakage equivalent. This COL information item is related to FSAR Subsection 5.2.5.9.

- COL License Information Item 5.3 Reactor Vessel Water Level Instrumentation

This COL license information item addresses the requirement for the applicant to design the reactor vessel water level instrumentation flow control system to provide adequate flow rates. This COL information item is related to FSAR Subsection 5.2.5.2.1.

5.2.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the Reactor Coolant Pressure Boundary and Core Cooling Systems Leakage Detection, and the associated acceptance criteria, are in Section 5.2.5 of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 1 and Tier 2 departures. Tier 1 departures requiring prior NRC approval are subject to the requirements specified in 10 CFR Part 52, Appendix A, Section VIII.A.4. Tier 2 departures affecting TS require prior NRC approval and are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.C.4. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

The regulatory basis for the review of the COL license information items is in Section 5.2.5 of NUREG–0800.

5.2.5.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 5.2.5 of the certified ABWR DCD. The staff reviewed Section 5.2.5 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic[7]. The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

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7 See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

The staff's review of this application includes the following considerations:

Tier 1 Departures

- STD DEP T1 2.3-1 — Deletion of MSIV Closure and Scram on High Radiation

The staff reviewed STP DEP T1 2.3-1, which involves the deletion of MSIV closure and reactor scram on high radiation. The staff finds that this departure is not relevant to the scope of review in SRP Section 5.2.5. This departure is evaluated in Chapter 7 of this SER.

- STD DEP T1 2.4-2 — Feedwater Line Break Mitigation

The staff reviewed STP DEP T1 2.4-2, which involves the addition of a trip of the condensate pumps upon indication that an FWLB in the drywell has occurred. The instrumentation logic used to initiate the condensate pump trip signal will be an "AND" circuit with inputs requiring an excessive differential pressure between the two feedwater lines and high-drywell pressure. The staff found that this departure is not relevant to the scope of review in SRP Section 5.2.5. This departure is evaluated in Chapter 7 of this SER.

Tier 2 Departure Requiring Prior NRC Approval

- STD DEP 7.3-12 — Leak Detection and Isolation System Sump Monitoring

The staff reviewed STD DEP 7.3-12, which describes modifications to the alarm setpoints to support TS limits for the RCPB. The total leakage limit, averaged over the previous 24-hour period changed from 95 L/min (25 gpm) to 114 L/min (30 gpm); the unidentified leakage limit changed from about 3.8 L/min (1 gpm) to 19 L/min (5 gpm); and a limit of unidentified leakage increase of about 7.6 L/min (2 gpm) within the previous 4-hour period while in Mode 1 was added. Subsection 5.2.5.9 of the Tier 2 FSAR states that the changes in total leakage limit and in the unidentified leakage limit satisfies Position C.9 in RG 1.45. However, it appeared that the applicant had not adequately justified this change in terms of meeting Regulatory Positions C.2 and C.5. The staff issued **RAI 05.02.05-1** requesting the applicant to justify the above changes in the TS limits regarding the RG 1.45 Regulatory Positions stated in C.2 (specifying a flow measurement accuracy of 3.785 L/min [1 gpm] or better) and C.5 (specifying sensitivity and response times for leakage detection of 3.785 L/min [1 gpm] in less than 1 hour).

The applicant's response to **RAI 05.02.05-1** dated June 26, 2008 (ML081970231), states that in NUREG–1503, pages 5 through 11, the NRC found that "The sensitivity and response time for all these primary detection systems is 3.79 L/min (1 gpm) or its equivalent in less than 1 hour, thus satisfying Positions C.2 and C.5 of RG 1.45, Revision 0." Furthermore, Departure STD DEP 7.3-12 does not make any changes to the sensitivity or response time for the primary detection systems. Consequently, NRC's finding that these systems satisfy Positions C.2 and C.5 in RG 1.45 Revision 0 remains valid and effective. The applicant also indicates that on this basis, there are no changes to the COL application required by this response.

The staff found that the applicant's response adequately addresses the staff's concerns in RAI 05.02.05-1. The changes made to the TS leakage limits do not change the accuracy, sensitivity, or response time capabilities of the leakage detection system. Therefore, **RAI 05.02.05-1** is closed.

In **RAI 05.02.05-2**, the staff stated the position that the alarm limit needs to be set as low as practicable to provide an early warning signal alerting the operator to take actions before the TS limit is reached. RG 1.45 (page 1.45-2) provides guidance on the "detector sensitivity" and states that, "sumps and tanks used to collect unidentified leakage and air cooler condensate should be instrumented to alarm for increases of from 0.5 to 1.0 gpm." The sensitivity of 3.785 L/min (1 gpm) claimed by STP is not demonstrated in the alarm setpoint or in the TS limit, and there is no explicit indication of use by operators under any procedures.

The applicant's response to **RAI 05.02.05-2** dated June 26, 2008 (ML081970231), states that the 8 L/min (2 gpm) within the previous 4-hour period limit that was added to the TS is intended to provide early warning to prevent violating the 19 L/min (5 gpm) and the 114 L/min (30 gpm) leakage rate limits. The applicant also states that this monitoring alarm setpoint is similar to the limits previously approved by the NRC for BWR/6 plants. The applicant states that they will revise the summary description of STD DEP 7.3-12 in Part 7 of the COL application to more clearly state the purpose of the "increase in unidentified leakage" parameter and to explain that the 8 L/min (2 gpm) increase in the 4-hour limit is a plant computer-based control room alarm that will provide early warning to the operators so they can take action well below the TS limit for the unidentified leakage rate of 19 L/min (5 gpm). The alarm is activated when an increase in leakage is above the normal leakage values.

The applicant has also stated that they will develop alarm response procedures to specify operator actions in response to unidentified leakage rates greater than the alarm setpoint and less than the TS limit. These procedures will instruct the operators to monitor available parameters and to initiate trending while the condition is investigated. These procedures are to be completed and available prior to fuel load. Based on the alarm setpoint and alarm response procedures, the staff determined that **RAI 05.02.05-2** is resolved. However, a commitment to develop these procedures only appears in the RAI response; and is not reflected in the FSAR. The commitment needs to be included in the FSAR as the basis for the safety finding. Therefore, **RAI 05.02.05-5** requested the applicant to clarify in the FSAR that there is a commitment to develop these procedures. This RAI is identified as **Open Item 05.02.05-5** in the SER with open items.

The operating experience at Davis Besse indicated that a prolonged low-level, unidentified leakage inside the containment could cause material degradation that could potentially compromise the integrity of a system and lead to the gross rupture of the RCPB. In **RAI 05.02.05-3**, the staff took issue with the STD DEP 7.3-12 change in the unidentified leakage limit from 3.785 L/min (1 gpm) to 19 L/min (5 gpm) on the basis that the applicant should establish a low-leakage alarm setpoint that is below the TS limit of 19 L/min (5 gpm), which would provide the operator sufficient time to take action before reaching the TS limit. Additionally, the staff indicated that the applicant should also establish procedures that specify operator actions in response to leakage rates that are less than the limits in the TS.

The applicant's response to **RAI 05.02.05-3** in a letter dated June 26, 2008 (ML081970231), refers to the applicant's response to RAI 05.02.05-2. In that response, STD DEP 7.3-12 adds a computer-based control room alarm set at an increase in an unidentified leakage of 8 L/min (2gpm) over the previous 4 hours. The applicant also states that this alarm will provide adequate early warning to the operators so they can take action well before the TS limit of 19 L/min (5 gpm) is reached. The applicant will revise Subsection 5.2.5.2.1 in the Tier 2 FSAR to clarify that STD DEP 7.3-12 adds an early warning of an RCS leakage to the operators by means of a computer-based control room alarm that requires operator action with an 8 L/min (2 gpm) increase in an unidentified leakage over four hours. The applicant also commits to establish procedures that will specify operator diagnostic and corrective actions to address the alarm. These procedures will be made available before fuel loading. The staff found this response acceptable, and **RAI 05.02.05-3** is therefore closed. However, a commitment to develop these procedures only appears in the RAI response; and is not reflected in the FSAR. The commitment needs to be included in the FSAR as the basis for the safety finding. Therefore, the staff issued **RAI 05.02.05-5** requesting the applicant to clarify in the FSAR that there is a commitment to develop these procedures. The applicant's response to **RAI 05.02.05-5** dated April 8, 2010 (ML101040253), refers to the response to RAI 05.02.05-3. In that response, the applicant revised FSAR Subsection 13.5.3.4.6, "Alarm Response Procedures," to add a clarifying statement that "included in this procedure group will be specific guidance specifying operator actions in response to prolonged low level reactor coolant system leakage below the Technical Specifications limits." Based on the revised FSAR statement, which addressed the concern identified in Open Item 05.02.05-5, the staff determined that **RAI 05.02.05-5** is resolved. The staff confirmed that the above changes are in COL FSAR Revision 4, and this RAI is therefore closed.

Tier 2 Departure Not Requiring Prior NRC Approval

The applicant has determined that the following Tier 2 departure does not require NRC review and approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5 requirements:

- STD DEP 7.3-11 — Leak Detection and Isolation System Valve Leakage Monitoring

NRC staff reviewed STD DEP 7.3-11, which describes modifications to the RCPB Leakage Detection System, specifically the elimination of piping and instrumentation for leakage detection from valve stems of large-bore reactor coolant pressure boundary isolation valves because of the use of expanded graphite valve stem seals. Section 3 of the Departures Report indicates that this departure was evaluated and determined to comply with the requirements in 10 CFR Part 52, Appendix A, Section VIII.B.5. STP DEP 7.3-11 will make any measure of identified leakage from the valve stem accounted for as part of an unidentified leakage measure. The unidentified leakage TS limit is more stringent than the identified leakage TS limit. Therefore, this approach is more conservative and does not cause any concerns with respect to the review criteria established in SRP Section 5.2.5.

The applicant's evaluation has determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that the departure does not require prior NRC approval. The applicant's process for evaluating departures to the certified ABWR DCD is subject to NRC inspections.

COL License Information Items

- COL License Information Item 5.1 — Conversion of Indicators

NRC staff reviewed COL License Information Item 5.1 in FSAR Subsection 5.2.6.1. The applicant's information addressing COL License Information Item 5.1 indicates that surveillance procedures will direct the operator to convert the drywell leakage indications into a common leakage equivalent to unidentified and identified leakages to ensure that leakage requirements in the TS are met. The surveillance procedure calls for measuring levels in various leakage collection tanks over prescribed time frames and converting these levels into a leakage rate. In **RAI 05.02.05-4,** the staff expressed concern with this methodology in the following areas:

- Only one of the four leakage detection instrumentations in plant TS LCO 3.4.5 was addressed. The instrumentation did not have the "various indicators" that are specified in the COL information item. The first paragraph under Section 5.2.5.1.1 of the ABWR DCD states, "*The primary detection method for small unidentified leaks within the drywell includes (1) drywell floor drain sump pump activity and sump level increases, (2) drywell cooler condensate flow rate increases, and (3) airborne gaseous and particulate radioactivity increases. The sensitivity of these primary detection methods for unidentified leakage within the drywell is 3.785 liters/min (1 gpm) within one hour. These variables are continuously indicated and/or recorded in the control room.*" Since this paragraph was incorporated by reference in the STP FSAR, the applicant should have specified how the rest of the various indications (i.e., drywell cooler condensate flow, airborne particulate and airborne gaseous radioactivity monitors) of an equivalent leakage would be established and provided to operations as part of the important parameters to be included in the surveillance procedures for determining leakage rates.

- The purpose of the COL procedures is not just limited to ensuring that the TS limits are met, but to also provide operators with leakage rate information to take actions in response to a low-level leakage.

- The applicant should address when the procedures will be available.

The applicant's response to **RAI 05.02.05-4** dated June 26, 2008 (ML081970231), states the intent to revise Subsection 5.2.6.1 in the Tier 2 FSAR to specify the four drywell leakage detection indications that will be addressed by the surveillance procedures (i.e., the drywell floor drain sump, drywell airborne particulate monitoring, drywell gaseous radioactivity monitoring, and drywell air cooler condensate flow). In addition, the procedures will address how the parameters will be converted to a common leakage equivalent for determining leakage rates. The applicant will complete the surveillance procedures and make them available before fuel load. Based on this response, the staff considered **RAI 05.02.05-4** closed. However, a commitment to develop these procedures only appears in the RAI response; and is not reflected in the FSAR. The commitment needs to be included in the FSAR as the basis for the safety finding. Therefore, the staff issued **RAI 05.02.05-5** requesting the applicant to clarify in the FSAR that there is a commitment to develop these procedures. This RAI was identified as **Open Item 05.02.05-5** in the SER with open items. The applicant's response to **RAI 05.02.05-5** dated April 8, 2010, revises FSAR Subsection 13.5.3.4.8, "Calibration, Inspection, and Test Procedures," to add a clarifying statement that "included in this procedure group will be guidance regarding the conversion of various leakage measurements into a common leakage equivalent." Based on this revised FSAR statement, which addresses the concern identified in **Open Item 05.02.05-5**, the staff determined that **RAI 05.02.05-5** is resolved**.** The staff confirmed that COL FSAR Revision 4 includes the proposed changes, and RAI 05.02.05-5 is therefore closed.

- COL License Information Item 5.3 Reactor Vessel Water Level Instrumentation

COL License Information Item 5.3 requires the COL applicant to design the reactor vessel water level instrumentation flow control system to provide flow rates determined by the results of the BWR Owners' Group testing.

The reactor vessel water level instrumentation backfill water flow is supplied from the CRD system to the reactor water level instrumentation leg to prevent the potential formation of a gas pocket in the reference leg. The impact of noncondensable gases on the accuracy of reactor vessel level measurements is considered in the system design. The CRD system provides a process flow of approximately 4 L/min based on the results of BWR Owners' Group Bulletin 9303. This new value will be confirmed during preoperational testing in accordance with FSAR Subsection 14.2.12.1.6(3)(d). The NRC staff found this acceptable and the requirements of the COL License Information Item 5.3 are satisfied.

5.2.5.5 *Post Combined License Activities*

To resolve Open Item 05.02.05-5, the applicant included in the FSAR clarifying statements to include specific guidance for operator actions in the following procedures before fuel loading: Alarm Response Procedures, and Surveillance (Calibration, Inspection, and Test) Procedures. These procedures are subject to inspection before fuel loading.

5.2.5.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to the RCPB leakage detection, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A ,Section VI.B.1, all nuclear safety issues relating to the RCPB leak detection that were incorporated by reference have been resolved.

On the basis of the review of information in the COL FSAR and the applicant's clarifications in the specified RAI responses, the staff concluded that the design of the systems and components for RCPB leakage detection is acceptable. The design meets the requirements of GDC 2 with respect to the capability of the systems and components to maintain and perform their safety functions in the event of an earthquake. The design also meets the requirements of GDC 30 with respect to the detection, identification, and monitoring of the source of a reactor coolant leakage. This conclusion is based on the following:

- The RCPB leakage detection design has fulfilled the requirements of GDC 2 with respect to the capability of systems and components to perform and maintain their safety functions in the event of an earthquake by meeting the guidelines in RG 1.29, Positions C.1 and C.2.

- The RCPB leakage detection design has fulfilled the requirements of GDC 30 with respect to the detection, identification, and monitoring of the source of a reactor coolant leakage by meeting the guidelines in RG 1.45, Positions C.1 through C.9.

Therefore, the staff concluded that RCPB leakage detection for the COL FSAR design conforms to the guidelines in SRP Section 5.2.5 and is thus acceptable.

The staff also found that the applicant has addressed the COL license information items, the TS, the ITAAC, and the Initial Test Program considerations related to this area of review.

5.3 <u>Reactor Vessel</u>

5.3.1 Reactor Vessel Materials

5.3.1.1 *Introduction*

This section of the FSAR addresses seven topic areas of RV materials: (1) material specifications, (2) special processes used to manufacture and fabricate components, (3) special methods for nondestructive examination, (4) special controls and special processes used for ferritic steels and austenitic stainless steels, (5) fracture toughness, (6) the RV materials Surveillance Capsule Program (RVSP), and (7) RV fasteners.

5.3.1.2 *Summary of Application*

Section 5.3.1 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.3.1 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in COL FSAR Section 5.3.1, the applicant provides the following:

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 5.3-1 Reactor Pressure Vessel Material Surveillance Program

This departure describes the RVSP that serves the purpose of ensuring that the RPV maintains its fracture toughness margins throughout the vessel lifetime.

- STD DEP Admin

The Admin departure is used to accomplish editorial changes that are required.

- STD DEP Vendor

This departure describes the applicant's decision to use the services of an alternate vendor to support the application.

COL License Information Items

- COL License Information Item 5.4 Fracture Toughness Data

This COL license information item establishes the requirement for the applicant to provide fracture toughness data in an amendment to the FSAR before the receipt of fuel on the site. (COM 5.3-1).

- COL License Information Item 5.5 Materials and Surveillance Capsule

This COL license information item requires the applicant to update to the COL FSAR before the receipt of fuel onsite to identify the specific materials in each surveillance capsule and provide a plant-specific replacement for the pressure-temperature limits. (COM 5.3-2).

5.3.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the reactor vessel materials, and the associated acceptance criteria, are in Section 5.3.1 of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 2 departures. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

The regulatory basis for the review of the COL license information items is in Section 5.3.1 of NUREG–0800. Specifically, the regulatory basis for acceptance of the COL information items is established in the following:

GDC 32 found in Appendix A to Part 50, as it relates to the RVSP.

10 CFR 50.60, as it relates to compliance with the requirements of 10 CFR Part 50, Appendix G.

10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness.

10 CFR Part 50, Appendix H, as it relates to the RVSP.

10 CFR 50.55a, as it relates to the requirements for testing and inspecting Class 1 components of the RCPB, as specified in Section XI of the ASME Code.

SECY-05-0197, as it relates to fully describing an operational program.

5.3.1.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 5.3.1 of the certified ABWR DCD. The staff reviewed Section 5.3.1 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[8] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the RV materials.

The staff reviewed the information in the COL FSAR:

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 5.3-1 Reactor Pressure Vessel Material Surveillance Program

In STD DEP 5.3-1, the applicant revises information incorporated by reference to ABWR DCD Subsection 5.3.1.6.1 and DCD Subsection 5.3.1.6.4. In STP Units 3 and 4 COL FSAR Subsection 5.3.1.6.1, "Compliance with Reactor Vessel Material Surveillance Program Requirements," the applicant proposes to remove the reference to GE Licensing Topical Report (LTR) NEDO–33315P, "Advanced Boiling Water Reactor (ABWR) Reactor Pressure Vessel (RPV) Materials Surveillance Program." NRC staff found that the proposed revision to

[8] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

Subsection 5.3.1.6.1 is in accordance with Appendix A, Section VIII.B.5 of 10 CFR Part 52. Therefore, this revision does not require prior NRC approval.

In STP Units 3 and 4 COL FSAR Subsection 5.3.1.6.4, “Position of Surveillance Capsules and Methods of Attachment Appendix H.II B (2),” the applicant proposes to revise the range of the surveillance capsule lead factors from “1.2 -1.5” to “1 to 1.5.” However, the proposed departure in COL FSAR Subsection 5.3.1.6.4 is not acceptable because ASTM E-185 recommends that the surveillance capsule lead factors be greater than one (1) and less than or equal to three (3). The staff issued **RAI 05.03.01-5** requesting the applicant to provide surveillance capsule lead factors that are in compliance with the recommendations of ASTM E–185, which therefore meet the requirements of Appendix H to 10 CFR Part 50. The applicant’s response to this RAI dated July 23, 2009 (ML092080080), states that the surveillance capsule specimen holders are located to produce a lead factor greater than one (1), which is in compliance with ASTM E–185, and the COL application will be revised to state that the lead factor is greater than one (1) and less than or equal to 1.5. The staff found that the applicant has appropriately responded to this RAI. In order to confirm that the proposed revision appears in an upcoming revision of the COL application, this RAI was tracked as **Confirmatory Item 05.03.01-1**. The staff confirmed in Revision 4 of the COL FSAR that the proposed change was made and therefore RAI 05.03.01-5 and Confirmatory Item 05.03.01-1 are closed. The staff’s review of the adequacy of the RVSP is discussed under COL License Information Item 5.5.

- STD DEP Admin

In STD DEP Admin, the applicant revises information incorporated by reference in STP Units 3 and 4 COL FSAR Subsection 5.3.1.6.1, “Compliance with Reactor Vessel Material Surveillance Program Requirements.” Specifically, this departure provides grammatical revisions (spelling, punctuation, etc.) to the referenced DCD text. This departure is editorial in nature.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval.

- STD DEP Vendor

STD DEP Vendor, the applicant revises information incorporated by reference in STP Units 3 and 4 COL FSAR Subsection 5.3.1.2, “Special Procedures Used for Manufacturing and Fabrication,” and Subsection 5.3.1.6.5, “Time and Number of Dosimetry Measurements.” Specifically, this departure proposes to remove all references to GE from the FSAR text.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval.

The staff’s review of this application includes the following considerations:

COL License Information Items

- COL License Information Item 5.4 Fracture Toughness Data

COL License Information Item 5.4 states that fracture toughness data based on the limiting RV materials will be provided. To address this COL license information item, STP Units 3 and 4 COL FSAR Subsection 5.3.4.1 states that the fracture toughness data based on the limiting RV actual materials will be provided in an amendment to the FSAR, in accordance with10 CFR 50.71(e) and before receiving fuels on the site. The data will be based on test results from the actual materials used in the RPV. The evaluation methods will be in accordance with Appendix G of the ASME BPV Code, Section III Division 1, 1989 Edition, excluding addenda; 10 CFR 50, Appendices G and H; and RG 1.99 Revision 2. NRC staff noted that the as-procured RV material properties will be available to the COL Holder after the acceptance of the RV. Therefore, in order to provide sufficient time for the NRC to review the fracture toughness data, the staff issued **RAI 05.03.01-1** requesting the applicant to commit to stating that within a reasonable period of time (i.e., 6 months or 1 year) following the acceptance of the RV, the COL Holder will submit the fracture toughness data to the staff for review. The applicant's response to this RAI dated April 2, 2009 (ML090960299), proposes to revise FSAR Subsection 5.3.4.1 to state that the fracture toughness data based on limiting RV actual materials will be provided in the first regular FSAR update issued 1 year after the onsite acceptance of the RV.(COM 5.3-1). The staff found that the applicant has appropriately addressed the RAI. However, the staff was tracking RAI 05.03.01-1 by **Confirmatory Item 05.03.01-2** to confirm that the proposed revision was included in an upcoming revision of the COL FSAR. The staff confirmed in Revision 4 of the FSAR that the proposed changes were included in the COL application. Therefore RAI 05.03.01-1 and Confirmatory Item 05.03.01-2 are closed.

- COL License Information Item 5.5 Materials and Surveillance Capsule

COL License Information item 5.5 states that the following will be identified: (1) the specific materials in each surveillance capsule; (2) the capsule lead factors; (3) the withdrawal schedule for each surveillance capsule; (4) the neutron fluence to be received by each capsule at the time of its withdrawal; and (5) the vessel end-of-life peak neutron fluence. In addition, RG 1.206 Chapter 5, C.I.5.3.1.6, "Material Surveillance," states that the RVSP and its implementation should be described in sufficient detail to ensure that the program meets the requirements of Appendix H to 10 CFR Part 50. RG 1.206 also lists the following topics that should be addressed in the description of the RVSP:

- Basis for the selection of material in the program
- Number and type of specimens in each capsule
- Number of capsules and proposed withdrawal schedule in compliance with the edition of ASTM E–185 Annual Book of ASTM Standards, Part 30, referenced in Appendix H to 10 CFR Part 50
- Neutron flux and fluence calculations for vessel wall and surveillance specimens
- Projected radiation embrittlement on vessel wall

- Location of capsules, method of attachment, and provisions to ensure that capsules are retained in position throughout the vessel lifetime

To address COL License Information Item 5.5 (COM-5.3-2), FSAR Subsection 5.3.4.2 provides the following site-specific description of the RVSP:

(1) Specific materials in each surveillance capsule

The surveillance specimens are fabricated from extra portions of vessel forging material from the core regions. The vessel material is low alloy steel, ASME SA-508 Class 3. Surveillance specimens are fabricated by sectioning a weldment made from the extra forging material. Surveillance specimens are taken from the base metal, weld metal and the heat affected zone of the weldment.

(2) Capsule lead factor

The lead factor of each capsule is approximately 1.1.

(3) Withdrawal schedule for each surveillance capsule

The capsule withdrawal schedule is in accordance with ABWR DCD Tier 2 Subsection 5.3.1.6.1.

(4) Neutron fluence to be received by each capsule at the time of its withdrawal

The neutron fluence to be received by each capsule is as follows:

(a) First capsule: 5.2×10^{16} n/cm^2

(b) Second capsule: 1.7×10^{17} n/cm^2

(c) Third capsule: not to exceed 5.0×10^{17} n/cm^2

(d) Fourth Capsule: will be based on the results of the first two capsules

(5) Vessel end-of life peak neutron fluence

The vessel end-of-life neutron fluence is approximately 5.0×10^{17} n/cm^2

The applicant also states that the RVSP for STP Units 3 and 4 is in accordance with the "STP 3 & 4 Reactor Pressure Vessel Surveillance Program," Toshiba Corporation, July 2008 (RS-5126528, Revision 1).

The staff issued **RAI 05.03.01-2** requesting the applicant to (a) provide the detailed locations of the surveillance capsules in the core beltline region, (b) describe in detail the process for preparing the capsule specimens, (c) specify the number and type of specimens in each capsule, and (d) provide the referenced document entitled "STP 3 & 4 Reactor Pressure Vessel Surveillance Program" for NRC review and approval. The applicant's response to RAI 05.03.01-2 dated April 2, 2009 (ML090960297), provides Topical Report UTLR-0003, Revision 0, "Reactor Vessel Materials Surveillance Program," Toshiba Corporation, April 2009 for NRC review. Upon review of the information provided in the Topical Report UTLR-0003, the staff found that the applicant has appropriately addressed RAI 05.03.01-2 by providing the detailed locations of the surveillance capsules in the core beltline region, the detailed process for preparing the capsule specimens, and the number and type of specimens in each capsule. In addition, the staff found that the STP Units 3 and 4 RVSP (UTLR-0003) is in accordance with ASTM E185 and therefore satisfies the requirements of Appendix H to 10 CFR Part 50. In supplemental **RAI 05.03.01-4,** the staff requested the applicant to revise FSAR Subsection 5.3.4.2,"Materials and Surveillance Capsule," and FSAR Section 5.3.5, "References," to reference Topical Report UTLR-0003 in the description of the "STP Units 3 and 4 Reactor Vessel Materials Surveillance Program." The applicant's response to this RAI dated July 23, 2009 (ML092080080), states that the COL application will be revised to update the reference. The staff found that the applicant has appropriately addressed the RAI. The staff tracked **Confirmatory Item 05.03.01-3** to confirm that the proposed revision is included in an upcoming revision of the COL application. The staff confirmed in Revision 4 of the FSAR that the proposed changes were included in the COL application. Therefore RAI 05.03.01-2, RAI 05.03.01-4 and Confirmatory Item 05.03.01-3 are closed.

The implementation milestones for the RVSP, and other operational programs, are provided in FSAR Section 13.4S. In COL FSAR Table 13.4S-1, the applicant proposes implementing the RVSP at initial criticality. However, in order for the staff to verify that the requirements of the RVSP have been met and are in accordance with Appendix H of 10 CFR Part 50, the program must be implemented before fuel loading. The staff issued RAI 05.03.01-3 requesting the applicant to revise the FSAR accordingly. The applicant's response to this RAI dated April 2, 2009 (ML090960299), states that the FSAR will be revised to clarify that the requirements of the RVSP will be met before fuel loading. The staff found that the applicant has appropriately addressed the RAI, which is being tracked as **Confirmatory Item 05.03.01-4.** The staff confirmed in Revision 4 of the FSAR that the proposed changes were included in the COL application. Therefore RAI 05.03.01-3 and Confirmatory Item 05.03.01-4 are closed.

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 staff expressed concerns about (1) the end-of-life Charpy upper-shelf energy predictions (USE) for end of life for the limiting beltline weld and the plate or forging; (2) RVs constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition; and (3) the 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.

Because there is only one opportunity (during vessel fabrication) to take the appropriate sample specimens from the actual material used in fabricating the beltline of the RV, the staff emphasized this issue in **RAI 05.03.01-2**. The staff asked the applicant to describe the process for preparing the capsule specimens.

The ABWR DCD states that the RV materials surveillance specimens are provided in accordance with the requirements of ASTM E–185. 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."

The staff found that by referencing the ABWR DCD and providing an appropriate response to **RAI 05.03.01-2**, the applicant has met the intent of GL 92-01. The applicant will continue to meet the intent of the GL in the future by providing summary test reports to the NRC.

5.3.1.5 *Post Combined License Activities*

The COL applicant will develop a plant-specific RVSP. The COL applicant states that a complete RVSP will be implemented before fuel loading by a license condition.

The applicant identifies the following commitments:

- Commitment COM 5.3-1 – Provide fracture toughness data in an amendment to the FSAR 1 year after onsite acceptance of the reactor vessel.

- Commitment COM 5.3-2 – Update the COL FSAR prior to the receipt of fuel onsite to identify the specific materials in each surveillance capsule and to provide a plant-specific replacement for the pressure-temperature limits.

5.3.1.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to the RV materials. No outstanding information is expected to be addressed in the COL FSAR related to this section[9]. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the RV materials that were incorporated by reference have been resolved.

The staff's review concluded that the applicant's proposed resolutions to COL License Information Items 5.4 and 5.5 meets NRC regulations and the relevant acceptance criteria of SRP Section 5.3.1 and the guidance in RG 1.206, Section CIII.1 Chapter 5, C.I.5.3.1.

5.3.2 Pressure-Temperature Limits

5.3.2.1 *Introduction*

Pressure-Temperature (P-T) limits are required as a means of protecting the reactor vessel during startup and shutdown to minimize the possibility of a fast fracture. The methods outlined in Appendix G of Section XI of the ASME Code are employed in the analysis of protection against a non-ductile failure. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference temperature (ART), which includes reference the nil ductility temperature (NDT) shifts, initial reference temperature $(RT)_{NDT}$, and margin.

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[9] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

5.3.2.2 *Summary of Application*

Section 5.3.2 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.3.2 of the certified ABWR DCD, Revision 4. In addition, in COL FSAR Section 5.3.2, the applicant provides the following:

<u>*COL License Information Item*</u>

- COL License Information Item 5.6 Plant Specific Pressure-Temperature Information

This COL license information item requires the applicant to submit plant-specific calculations of RT_{NDT}, stress intensity factors, and P-T limit curves. (COM 5.3-3).

5.3.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the P-T limits, and the associated acceptance criteria, are in Section 5.3.2 of NUREG–0800.

The regulatory basis for acceptance of the resolution to the COL License Information Item 5.6 is Appendix G to 10 CFR Part 50, as it relates to fracture toughness requirements.

5.3.2.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 5.3.2 of the certified ABWR DCD. The staff reviewed Section 5.3.2 of the STP Units 3 and 4 COL FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[10] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the P-T limits.

The staff's review of this application includes the following considerations:

<u>*COL License Information Item*</u>

- COL License Information Item 5.6 Plant Specific Pressure-Temperature Information

COL License Information Item 5.6 states that the COL applicant will submit plant-specific, P-T limit curves. To address this COL license information item, FSAR Subsection 5.3.4.3 states that plant-specific, P-T limit curves developed using Appendix G of ASME Code Section XI will be provided in an amendment to the FSAR, in accordance with 10 CFR 50.71(e) before receiving fuel onsite (COM 5.3-3). The staff issued **RAI 05.03.02-1** requesting the applicant to provide P-T limits or a P-T limits report (PTLR) for STP Units 3 and 4 to the NRC before the issuance of a COL license and to provide either the P-T limits or the PTLR for NRC review and approval. This item was tracked as **Open Item 05.03.02-1** in the SER with open items**.** In a letter dated July 23, 2009 (ML092080079), the applicant submitted Technical Report

[10] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

U7-C-STP-NRC-090080, "South Texas Project (STP) Units 3 & 4 Pressure-Temperature Limits Report (PTLR) and Fluence Calculation Methodology," for NRC approval. As documented in the safety evaluation dated October 5, 2010 (ML102660658), the staff has reviewed the PTLR and approved its use for the STP Units 3 and 4 reactor vessels for establishing limiting P-T limit curves and related input parameters. In addition, the STP 3 & 4 R-COL Technical Specifications (TSs) contain all of the necessary provisions required for the implementation and control of a PTLR. The relevant TS requirements include the TS definition of the PTLR (TS Section 1.1); the TS Limiting Conditions of Operation (LCO) for the Reactor Coolant System (RCS) P-T limits (LCO 3.4.9), including LCO Action Statements, Surveillance Requirements, and related applicability criteria; and the necessary administrative controls governing the PTLR content and reporting requirements (TS 5.7.1.6). All of the TS pages related to the implementation and control of a PTLR are acceptable to the staff. Based on this evaluation of the PTLR, **Open Item 05.03.02-1** is resolved**.**

5.3.2.5 *Post Combined License Activities*

The COL applicant will update the plant-specific, P-T limits using the PTLR methodology and inform the NRC of the updated P-T limits. No further review is needed if the PTLR methodology remains unchanged.

The applicant identifies the following commitment:

- Commitment (COM 5.3-3) – Provide an amendment to the FSAR regarding pressure-temperature curves before the receipt of fuel on site.

5.3.2.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to the P-T limits, and no outstanding information is expected to be addressed in the COL FSAR related to this section.[1] Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the P-T limits that were incorporated by reference have been resolved.

The staff's review concluded that the applicant's proposed resolution to COL License Information Item 5.6 meets 10 CFR Part 50, Appendix G, and the relevant acceptance criteria of SRP Section 5.3.2.

5.3.3 Reactor Vessel Integrity

5.3.3.1 *Introduction*

This section of the FSAR describes the RV integrity. The ABWR RV is a vertical cylindrical pressure vessel of welded construction. The cylindrical shell, top head, and bottom head of the RV are fabricated from low-alloy steel. The interior of the RV is clad with a stainless steel overlay. However, the top head, all of the nozzles (excluding the steam outlet nozzles), and the reactor internal pump (RIP) casings do not have cladding. The bottom head is clad with Ni-Cr-Fe alloy. The RIP penetrations are clad with Ni-Cr-Fe alloy or alternatively, with stainless steel.

5.3.3.2 *Summary of Application*

Section 5.3.3 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.3.3 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in COL FSAR Section 5.3.3, the applicant provides the following:

<u>*Tier 1 Departure*</u>

- STD DEP T1 2.1-2 Reactor Pressure Vessel System RIP Motor Casing Cladding

This departure revises information in ABWR DCD Tier 1, Section 2.1.1, "Reactor Pressure Vessel System," and DCD Tier 2 Subsection 5.3.3.1.1.1, "Reactor Vessel." Specifically, the departure modifies the description of RIP motor casing to clearly indicate that some portions of the motor casing have cladding.

<u>*Tier 2 Departures Not Requiring Prior NRC Approval*</u>

- STD DEP Vendor

This departure removes all references to GE from the FSAR text.

- STD DEP Admin

This departure proposes grammatical revisions (e.g., spelling and punctuation) to the referenced DCD text.

5.3.3.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the RV integrity and the associated acceptance criteria are in Section 5.3 of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures" of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 1 and Tier 2 departures. Tier 1 departures require prior NRC approval and are subject to the requirements of 10 CFR Part 52 Appendix A, Section VIII.A.4. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

5.3.3.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 5.3.3 of the certified ABWR DCD. The staff reviewed Section 5.3.3 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[11] The staff's review confirmed that the

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[11] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

information in the application and the information incorporated by reference address the required information relating to the RV integrity

The staff reviewed Section 5.3.3 of the STP Units 3 and 4 COL FSAR and the corresponding section of the ABWR DCD. Specifically, the staff reviewed Section 5.3.3 of the ABWR DCD to ensure that the information is appropriate for incorporation by reference, and any supplemental information to be provided has been addressed in the COL application.

The staff also reviewed the conformance of Section 5.3.3 of the STP COL FSAR to RG 1.206 Section C.III.1, Chapter 5, C.I.5.3.3, "Reactor Vessel Integrity."

The staff reviewed the information in the COL FSAR:

<u>Tier 1 Departure</u>

- STD DEP T1 2.1-2 Reactor Pressure Vessel System RIP Motor Casing Cladding

ABWR DCD Tier 2 Subsection 5.3.3.1.1.1 and DCD Tier 1 Section 2.1.1 state that the RIP casings do not have cladding. In this departure, the applicant clarifies that some portions of the RIP motor casings have cladding. STP Units 3 and 4 COL FSAR Tier 1 Section 2.1.1, "Reactor Pressure Vessel System," and Tier 2 Subsection 5.3.3.1.1.1, "Reactor Vessel," state that the RIP motor casings are clad with stainless steel only in the stretch tube region and around the bottom of the RIP motor casings. However, in the STP Units 3 and 4 COL application Part 7,"Departures Report," Section 2.1, STD DEP T1 2.1-2, the applicant states that the standard ABWR design for installed applications includes stainless steel cladding from the top portion of the casing to the motor secondary seal and around the bottom of the RIP motor casing. The applicant also states that this change represents an improvement based on the ABWR operating experience. The staff issued **RAI 05.03.03-1** requesting the applicant to (1) clarify which portions of the motor casing will have cladding, (2) provide the specific operating experience used to justify the design change, and (3) provide details about the material types and procedures used to join the RIP motor casing and the bottom reactor vessel head. The applicant's response to this RAI dated September 14, 2009 (ML092580477), provides a diagram and a description of the RIP motor casing materials, cladding, and the full penetration weld used to join the motor casing to the bottom head. The applicant also states that this design is currently used in the K-6 (Kashiwazaki Kariwa Nuclear Power Generation Station, Unit 6) and H-5 (Hamaoka Unit 5) ABWR units that have been operating in Japan since 1996 and 2005, respectively. In these plants, the RIPs have performed properly. In addition, the applicant proposes to revise COL application Part 7, STD DEP T1 2.1-2 to state that the RIP motor casings are clad with stainless steel only in the stretch tube region and around the bottom of the RIP motor casings. This change is consistent with the description in COL FSAR Tier 1 Section 2.1.1 and Tier 2 Subsection 5.3.3.1.1.1. The staff found that the applicant has appropriately responded to the RAI, which was tracked as **Confirmatory Item 05.03.03-1** to confirm that the proposed revision is included in the upcoming revision of the COL application. The staff confirmed in revision 4 of the FSAR that the proposed changes were included in the COL application. Therefore RAI 05.03.03-1 and Confirmatory Item 05.03.03-1 are closed.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP Vendor

The departure removes all references to GE from the FSAR text. The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval.

- STD DEP Admin

The departure proposes grammatical revisions (e.g., spelling and punctuation) to the referenced DCD text. The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval.

5.3.3.5 *Post Combined License Activities*

There are no post COL activities related to this section.

5.3.3.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to the RV integrity. No outstanding information is expected to be addressed in the COL FSAR related to this section.[12] Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the RV integrity that were incorporated by reference have been resolved.

The staff also concluded that STD DEP T1 2.1-2 meets the relevant acceptance criteria in Section 5.3 of NUREG–0800 and NRC regulations, and is thus acceptable. The staff found it reasonable that the identified Tier 2 departures do not require prior NRC approval per 10 CFR Part 52, Appendix A, Section VIII.B.5.

5.4 Component and Subsystem Design

5.4.1 Reactor Recirculation System

5.4.1.1 *Introduction*

This section of the FSAR addresses the ABWR reactor recirculation system, which is unique compared to existing BWR plants, because the reactor recirculation pumps are internal to the reactor vessel and the external piping loops have been eliminated. There are 10 RIPs in the ABWR design.

[12] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

5.4.1.2 *Summary of Application*

Section 5.4.1 of the STP Units 3 and 4 incorporates by reference Section 5.4.1 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52 Appendix A. In addition, in FSAR Section 5.4.1, the applicant provides the following:

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 5.4-2 Reactor Internal Pump (RIP) Motor Cable Box

This departure revises the RIP cross-sectional illustration, Figure 5.4-1, by reducing the size of the cable box and showing a plug-in type bower connector.

- STD DEP 5.4-4 Recirculation Motor Cooling Subsystem

This departure permits the fabrication of the recirculation motor heat exchanger (RMHX) shell, tube, tubesheet, and water box material using carbon steel or stainless steel.

COL License Information Item

- COL License Information Item 5.10 RIP Installation and Verification During Maintenance

The applicant provides information to address COL License Information Item 5.10 from the generic DCD. The applicant states that procedures will be developed to perform RIP installation and verification for the motor bottom cover, as well as visual monitoring of the potential leakage during impeller-shaft and plug removal. In addition, a contingency plan will assure that core and spent fuel cooling can be provided in the event of loss of coolant during RIP maintenance.

5.4.1.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the reactor circulation system, and the associated acceptance criteria, are in Section 5.4 of NUREG–0800.

In accordance with Section VIII, “Processes for Changes and Departures,” of “Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor,” the applicant identifies Tier 2 departures. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

The regulatory basis for the review of the COL license information items is in Section 5.4 of NUREG–0800.

5.4.1.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 5.4.1 of the certified ABWR DCD. The staff reviewed Section 5.4.1 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[13] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the reactor recirculation system.

The staff's review of this application includes the following considerations:

<u>*Tier 2 Departures Not Requiring Prior NRC Approval*</u>

- STD DEP 5.4-2 Reactor Internal Pump (RIP) Motor Cable Box

This departure revises the RIP cross-sectional illustration, Figure 5.4-1, by reducing the size of the cable box and showing a plug-in type bower connector. This change will improve maintainability and has no effect on RIP operation or performance. The RIP motor case box and plug-in connector are nonsafety-related components. The proposed departure is acceptable.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that the departure does not require prior NRC approval. The applicant's process for evaluating departures to the certified ABWR DCD is subject to NRC inspections.

- STD DEP 5.4-4 Recirculation Motor Cooling Subsystem

ABWR DCD Tier 2, Subsection 5.4.1.3.1 identifies the RMHX shell, tube, sheet, and water box material such as carbon steel. This departure permits the fabrication of these components using carbon steel or stainless steel. STD DEP 5.4-4 modifies Subsection 5.4.3.1 by adding stainless steel as a material that may be used to fabricate these components. STP COL Application Part 7, Section 3.0, "Departures Not Requiring NRC Approval," provides a summary of the applicant's evaluation of the addition of stainless steel for the fabrication of these components. NRC staff agrees that the use of stainless steel is acceptable for the intended application. It is considered an improvement in material selection over carbon steel, because stainless steel is resistant to flow accelerated corrosion.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. The staff found that the applicant has provided an adequate summary of this modification in the Departure Report and the modification meets the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that the departure does not require prior NRC approval. The applicant's process for evaluating departures and other changes to the DCD is subject to NRC inspections.

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13 See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

COL License Information Item

- COL License Information Item 5.10 RIP Installation and Verification During Maintenance

To comply with COL License Information Item 5.10, the applicant will develop specific procedures and a contingency plan to address activities related to RIP maintenance. The procedures will address (1) the RIP installation, (2) verification of the RIP motor bottom cover, and (3) visual monitoring of potential leakage during impeller shaft and maintenance removal. The contingency plan will provide instructions for activities that will ensure the availability of core and spent fuel pool cooling in the event that a loss of coolant occurs during RIP maintenance. Because some RIP maintenance activities have the potential to drain the reactor vessel, the staff required additional information from the applicant in **RAI 05.04-1** regarding the contingency plan.

Upon completion of the contingency plan described in COL License Information Item 5.10, **RAI 05.04-1** requested the applicant to provide the contingency plan for NRC staff to review, as it relates to the potential draining of the reactor vessel during RIP maintenance activities:

- Worst-case scenario evaluation
- Impact on personnel and plant
- Assumptions made
- Response time of plant and personnel
- Worst-case flow rate of the vessel draindown
- Number of pumps the plant procedures allow to perform concurrent maintenance activities that have the potential to drain the vessel
- Recovery phase

The applicant’s response to RAI 05.04-1 dated July 2, 2009 (ML091880282), refers to FSAR Subsection 13.5.3.3.1, which discusses administrative procedures. According to the applicant, these procedures will be developed only 6 months before pre-operational testing. However, the applicant does not address the RAI with respect to the contingency plan related to the specifics stated in the RAI. Therefore, the applicant’s response is not acceptable. The staff issued **RAI 05.04-2** requesting the applicant to address RAI 05.04-1 upon completion of the contingency plan. In the response to this RAI dated September 24, 2009 (ML092710223), the applicant agrees to include the description of the contingency plan in the FSAR, which will be included in the administrative procedures involving RIP maintenance. The staff verified that FSAR Subsection 5.4.15.4 of COL, FSAR Revision 4 includes the contingency plan. Therefore, **Open Item 05.04-2** is closed.

5.4.1.5 *Post Combined License Activity*

There are no post COL activities related to this section.

5.4.1.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to the reactor recirculation system, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the reactor recirculation system that were incorporated by reference have been resolved.

The staff reviewed the STP Units 3 and 4 COL application with respect to the relevant NRC regulations, the acceptance criteria in NUREG–0800 Section 5.4, and other NRC regulatory guidance. The staff found it reasonable that the identified Tier 2 departures are characterized as not requiring prior NRC approval per 10 CFR Part 52, Appendix A, Section VIII.B.5. The staff determined that the applicant is in compliance with the NRC regulations.

5.4.2 Steam Generator

This section is not applicable to the ABWR.

5.4.3 Reactor Coolant Piping

Section 5.4.3 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.3, "Reactor Coolant Piping," of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[14] The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the reactor coolant piping have been resolved.

5.4.4 Main Steamline Flow Restrictors

Section 5.4.4 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.4, "Main Steamline Flow Restrictors," of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the main steamline flow restrictors that were incorporated by reference have been resolved.

5.4.5 Main Steamline Isolation System

5.4.5.1 *Introduction*

This section of the FSAR addresses the operation of the MSIVs in the ABWR design. The ability to isolate the main steamlines provides the capability to limit the release of reactor coolant outside the containment, in the event of a steamline break.

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14 See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

5.4.5.2 *Summary of Application*

Section 5.4.5 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.5 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 5.4.5, the applicant provides the following:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 10.1-3 Rated Heat Balance

STP FSAR Subsection 5.4.5.2, "Description," references Departure STP DEP 10.1-3, "Rated Heat Balance," and provides a small adjustment to the rated steam flow through each MSIV.

COL License Information Item

- COL License Information Item 5.7 Testing of Main Steam Isolation Valves

ABWR DCD Tier 2 Subsection 5.4.15.1, "Testing of Main Steam Isolation Valves," specifies in COL License Information Item 5.7 that COL applicants will test the steam isolation valves in actual operating conditions. In response to this COL license information item, STP FSAR Subsection 5.4.15.1 states that testing the MSIVs under operating conditions will be performed during the Initial Test Program, as described in Subsection 14.2.12.2.26, "MSIV Performance," and Subsection 14.2.12.2.34, "Reactor Full Isolation." STP FSAR Subsections 14.2.12.2.26 and 14.2.12.2.34 provide minor edits to these DCD subsections.

STP FSAR Subsection 5.4.15.1 states that ITAAC Item 6 in ABWR DCD Tier 1 Table 2.1.2, "Nuclear Boiler System," will ensure that the MSIVs meet their design basis. The design commitment in ITAAC Item 6 specifies that the MSIVs are capable of closing within 3 to 4.5 seconds under differential pressure, fluid flow, and temperature conditions. The inspections, tests, and analyses in ITAAC Item 6 specify that tests of the as-built MSIVs will be conducted under preoperational differential pressure, fluid flow, and temperature conditions; and tests or type tests of an MSIV will be conducted under design-basis differential pressure, fluid flow, and temperature conditions.

5.4.5.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, NRC staff reviewed STP FSAR Section 5.4.5 using the review procedures described in Section 5.4, "Reactor Coolant System Component and Subsystem Design," of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 2 departures. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

5.4.5.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 5.4.5 of the certified ABWR DCD. The staff reviewed Section 5.4.5 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[15] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the main steam isolation system.

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 10.1-3 Rated Heat Balance

The staff's review of Departure STP DEP 10.1-3 and the rated steam flow adjustment is in a separate section of this SER. The staff addressed the processes for the functional design, qualification, and IST Programs for safety-related valves including the MSIVs, for STP Units 3 and 4 in Section 3.9.6 of this SER.

COL License Information Item

- COL License Information Item 5.7 Testing of Main Steam Isolation Valves

The COL applicant addresses COL License Information Item 5.7 related to MSIV testing in ABWR DCD Tier 2 Subsection 5.4.15.1, as follows:

- COL applicants will test the steam isolation valves under actual operating conditions (6.87 MPa gauge [G], 286 °C).

- STP FSAR Section 5.4.15.1 states that testing of the MSIVs under operating conditions will be performed during the Initial Test Program, as described in Subsections 14.2.12.2.26 and 14.2.12.2.34. The STP FSAR also states that ITAAC Item 6 in ABWR DCD Tier 1, Table 2.1.2, will ensure that the MSIVs meet their design basis.

NRC staff found that the COL applicant has adequately addressed the testing of the STP MSIVs through the processes established for the Functional Design, Qualification, Initial Testing, and IST Programs for these valves.

5.4.5.5 *Post Combined License Activities*

There are no post COL activities related to this subsection.

5.4.5.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. No outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety

[15] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

issues relating to the main steam isolation system that were incorporated by reference have been resolved.

The staff reviewed the STP Units 3 and 4 COL application with respect to the relevant NRC regulations, acceptance criteria in NUREG–0800 Section 5.4, and other NRC regulatory guidance. The staff determined that the applicant is in compliance with the NRC regulations.

5.4.6 Reactor Core Isolation Cooling System

5.4.6.1 *Introduction*

This section of the FSAR addresses the reactor core isolation cooling (RCIC) system. The RCIC is a safety system that serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. Unlike the previous generation of BWR plants, the RCIC in the ABWR is part of the emergency core cooling system. The ABWR RCIC system also provides the decay heat removal necessary for coping with a station blackout (SBO).

5.4.6.2 *Summary of Application*

Section 5.4.6 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.6 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in COL FSAR Section 5.4.6, the applicant provides the following:

Tier 1 Departure

- STD DEP T1 2.4-3 RCIC Turbine/Pump

In STP Units 3 and 4 FSAR Section 5.4.6, the applicant submits Tier 1 Departure STD DEP T1 2.4-3 RCIC Turbine/Pump. This departure involves the replacement of the RCIC turbine and pump system design with an integrated (monoblock) alternate turbine-pump system design. Tier 1 Section 2.4.4 discusses the departure and the simplification of the RCIC system. The simplifications are the removal of unnecessary components such as the barometric condenser, the vacuum pump, the condensate pump, valves, and associated equipment. Tier 2 Subsection 5.4.6.2.1.3 discusses interlocks and the removal of the valves. Tier 2 Subsection 5.4.6.2.2.1 discusses design condition changes related to the simplification of the RCIC design. Subsection 5.4.6.2.5.2 discusses the emergency mode and the changes related to Departure T1 2.4-3.

COL License Information Item

- COL License Information Item 5.8 Analyses of 8-hour RCIC Capability

The applicant commits (COM 5.4-1) to address the capability of the RCIC system to operate for 8 hours and provide a best estimate analysis for NRC review by the end of preoperational testing, thus demonstrating that the RCIC system can function for 8 hours in an SBO event.

Furthermore, the applicant commits (COM 5.4-2) to complete a best estimate analysis demonstrating that an adequate direct current (DC) battery and pneumatic supply capacity based on the as-purchased equipment configuration and make it available for NRC review before the commencement of the Preoperational Test Program.

5.4.6.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the RCIC system, and the associated acceptance criteria, are in Section 5.4.6 of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 1 departures. Tier 1 departures requiring prior NRC approval are subject to the requirements specified in 10 CFR Part 52, Appendix A, Section VIII.A.4.

The regulatory basis for reviewing the COL license information items is in Section 5.4.6 of NUREG–0800.

5.4.6.4 *Technical Evaluation*

As documented in NUREG–1503, the staff reviewed and approved Section 5.4.6 of the certified ABWR DCD. The staff reviewed Section 5.4.6 of the STP Units 3 and 4 COL FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[16] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the RCIC system.

The staff reviewed the information in the COL FSAR:

<u>*Tier 1 Departure*</u>

- STD DEP T1 2.4-3 RCIC Turbine/Pump

In STP Units 3 and 4 FSAR Section 5.4.6, the applicant identifies Tier 1 Departure STD DEP T1 2.4-3. This departure involves the replacement of the RCIC turbine and pump system design with an integrated (monoblock) alternate turbine-pump system design. Tier 1 Section 2.4.4 discusses the departure and the simplification of the RCIC system. The simplifications are the removal of unnecessary components such as the barometric condenser, the vacuum pump, the condensate pump, valves, and associated equipment.

The following changes were made to Tier 1 Section 2.4.4:

- The barometric condenser was deleted from the RCIC System. This change is acceptable because there is no barometric condenser in the new Turbine design.

- In Table 2.4.4, "Reactor Core Isolation Cooling System," Items 3c, 3e, and 3f were revised. The steam supply bypass valve logic description and the 10-second time delay signal were deleted from the acceptance criteria. These changes are

16 See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

acceptable because there is no steam supply bypass valve or 10-second timer with the new turbine design.

- Also in Table 2.4.4, Item 3i was revised. The pump torque was deleted from the acceptance criteria. This change is acceptable because this parameter cannot be directly measured in the integrated turbine/pump configuration.

- In Figure 2.4.4a, “Reactor Core Isolation Cooling System,” the steam supply bypass valve, Note 1, and class barrier were deleted. The steam supply bypass valve was deleted because there is no steam supply bypass valve with the new turbine design. Hence, this change is acceptable. The deletions of Note 1 and the class barrier from the figure are acceptable because the new turbine will be qualified according to ASME Code Section III.

Tier 2 Subsection 5.4.6.2.1.3 discusses interlocks and the removal of the valves. Tier 2 Subsection 5.4.6.2.2.1 discusses design condition changes related to the simplification of the RCIC design. Subsection 5.4.6.2.5.2 discusses the Emergency Mode and the changes related to Departure T1 2.4-3.

The following changes were made to Tier 2 Section 5.4.6:

- In FSAR Subsection 5.4.6.2.1.3, “Interlocks,” the interlocks to valves FO47, FO45, FO31, FO32, turbine trip and throttle valve (Part of COO2) and FO12 were deleted. These changes are acceptable because, these valves are not needed due to the elimination of the support systems for the new turbine/pump design.

- In FSAR Subsection 5.4.6.2.2.1, “Design Conditions,” Item 2 was deleted and Items 1 and 3e were revised. These changes are acceptable because the revisions reflect the new integrated turbine design.

- FSAR Subsection 5.4.6.2.5.2, “Emergency Mode (Transient and LOCA Event),” was revised to describe the new design. The new turbine/pump set is an integrated component, and the RCIC system utilizes a flow control system that is an integrated part of the pump and turbine. The proposed changes are therefore acceptable.

- In FSAR Table 5.4-1a, “Net Positive Suction Head (NPSH) Available to RCIC Pump,” the applicant changed the NPSH calculation due to the new cassette-type strainer used in the ECCS pumps. The pump NPSH margin changed from 0.35m to “2.84m-($H_F + H_{ST}$),” in which H_F is the maximum frictional head excluding strainer frictional head and H_{ST} is the frictional head. Revision 3 of the COL FSAR Table 5.4.1a states that the “final system design will meet the required NPSH with adequate margin.”

  The applicant submitted the technical bases for the NPSH change in response to **RAI 05.04.06-1** in a letter dated July 2, 2009 (U7-STP-NRC-090062). The response did not provide the pump NPSH margin and hence, this issue is not resolved. NRC staff issued **RAI 05.04.06-3** as a supplemental RAI that states, “A new ECCS suction strainer design was incorporated on STP 3 & 4 (STD DEP 6C-1) with a cassette type strainer. The symbols H_F and H_{ST} were provided without numerical values because the new strainer head loss had not been determined. The applicant needs to submit the results of the pump NPSH calculations showing the available NPSH margin when the new strainer head loss is determined." This RAI was tracked as **Open Item 05.04.06-3** in the SER with open items.

The applicant's response to RAI 05.04.06-3 dated June 22, 2010 (ML101750069), provided the maximum frictional head including strainer as 2.10 m of water, in which is same as the value provided in the ABWR DCD. However, the staff was unable to verify this value because the applicant deleted a table in the FSAR containing the RCIC suction strainer area. Therefore, on May 10, 2011 (ML111370046), the staff performed an audit of the applicant's documentation containing the RCIC suction strainer design information . During this audit the staff verified that the STP Units 3 and 4 RCIC pump suction strainers have an area which is several times larger than those described in the ABWR DCD. Based on larger strainer area and that the applicant replaced the fiber insulation in the ABWR certified design with reflective metallic insulation, which would give a significantly lesser head loss than fiber insulation, the staff determined that the maximum frictional head of 2.10 m of water was acceptable.

In response to a question asked during a meeting with Advisory Committee on Reactor Safeguards on March 8, 2011, the applicant in its letter U7-C-NINA-NRC-110053 dated April 5, 2011 submitted additional information with regard to the assumption of 100°C for the suppression pool temperature in the RCIC pump NPSH calculations. The value for static head of water was calculated incorrectly assuming 100°C instead of 77°C, the maximum suppression pool temperature during RCIC operation. FSAR Table 5.4-1a was corrected assuming the value for 77°C. ABWR DCD Vapor pressure value of 4.33m was corrected to 4.39m for 77°C. Also, the incorrect ABWR DCD atmospheric value of 10.26 m used in the NPSH available calculation was corrected to 10.62 m in the new NPSH available calculation. The revised pump NPSH calculation shown in the attached FSAR Table 5.4-1a indicates a margin of 0.59m and it is acceptable. In the ABWR DCD Table 5.4-1a, the NPSH margin is 0.39m.

The pump NPSH margin will be verified by the ITAAC as shown in ITAAC Table 2.4.4, item J. Because there will be a sufficient NPSH margin for the RCIC pump, the staff considered the issue resolved and **Open Item 05.4.06-3** is therefore closed. The proposed revision to FSAR Table 5.4-1.a will be tracked as **Confirmatory Item 05.4.06-3**.

- In FSAR Table 5.4.2, "Design Parameters for RCIC System Components," the applicant removed the cooling water flow from Item (1). This change is acceptable because the new design does not need cooling water flow. Also, the total pump discharge flow was deleted because this parameter is a duplicate of the injection flow. Item (3) was deleted because there are no leak-off orifices in the new design. The pump NPSH requirement was changed from 7.3 m to 7 m due to the introduction of the new strainer design. Item (4), valve operation requirements for valves F012, FO13, FO31, FO32, FO45, FO46, and FO47 was deleted because these valves listed in the DCD are no longer applicable to the new system design. These changes are all due to the new turbine design change and are acceptable.

ABWR DCD Tier 2, Section 5.4.6 describes the RCIC system for the ABWR design certified in 10 CFR Part 52. STP FSAR Section 5.4, “Components and Subsystem Design,” incorporates by reference ABWR DCD Tier 2 Section 5.4.6, but specifies Departure STD DEP T1 2.4-3 for a new RCIC turbine-pump design. The STP FSAR reflects the new RCIC turbine-pump design in several sections such as Section 3.9, “Mechanical Systems and Components”; Table 3.9-8, “Inservice Testing Safety-Related Pumps and Valves”; Section 3.9.6, “Testing of Pumps and Valves”; and Subsection 6.3.2.2.3, “Reactor Core Isolation Cooling System (RCIC).” STP COL Application Part 9, “Inspections, Tests, Analyses, Acceptance Criteria [ITAAC],” revises the ITAAC to reflect the design change to an integrated RCIC turbine and pump.

On April 29, 2009, NRC staff conducted an onsite review of supporting documentation describing the new RCIC turbine-pump design at the Westinghouse office in Rockville, Maryland (MD). The applicant’s response to **RAI 05.04.06-2-1** dated July 7, 2009 (ML092190224), submitted proprietary Toshiba Technical Report UTLR-0004-P (Revision 0, June 2009), “Application of Turbine Water Lubricated (TWL) Pump to South Texas Project Unit 3 & 4 RCIC Turbine-Pump.” This response addresses this RAI and other RAIs on the new RCIC turbine-pump design. On November 10 and 11, 2009, NRC staff conducted an audit of documentation supporting the STP COL application at the Westinghouse office in Rockville, MD. The staff followed the guidance in the Office of New Reactors (NRO) Office Instruction, NRO-REG-108, “Regulatory Audits,” in performing the audit. One audit objective was to review information that supports the description of the new RCIC turbine-pump design to be developed for STP Units 3 and 4. The audit results are summarized in an NRC letter dated December 7, 2009 (ADAMS Accession No. ML093220094). For example, the NRC staff found that Revision 0 to Toshiba Technical Report UTLR-0004-P did not include provisions for the functional qualification of the new RCIC turbine pump. As a follow-up action after the November 2009 audit, the applicant indicated its plan to submit a revision to the RCIC turbine pump topical report to specify the functional qualification provisions for the RCIC turbine pump. Therefore, the NRC staff determined that this RAI would be tracked as **Open Item 05.04.06-1**.

On May 10, 2010 (ML101320258), the applicant submitted Revision 1 (dated March 2010) of the Toshiba Technical Report UTLR-0004-P in response to the audit findings. Section 9.3, “Qualification Information,” in Toshiba Technical Report UTLR-0004-P (Revision 1) specifies that the TWL pump will be functionally qualified to perform its required functions in compliance with ASME Standard QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” as accepted in Revision 3 to RG 1.100, “Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants.” ASME Standard QME-1–2007 incorporates lessons learned from nuclear power plant operating experience and research programs for qualifying nuclear power plant mechanical equipment. This standard is accepted for use in Revision 3 to RG 1.100 with certain staff positions. The staff finds the reference to ASME Standard QME-1–2007 acceptable as it is addressed in Revision 3 to RG 1.100 for the functional qualification of the RCIC turbine pump. Therefore, **RAI 05.04.06-2-1** is resolved and **Open Item 05.04.06-1** is closed.

The RCIC turbine-pump design includes a small pump that returns leak-off water to the main pump suction lines. The staff issued **RAI 05.04.06-2-2** requesting the applicant to discuss this leak-off pump and its design, qualifications, and IST provisions. The applicant’s response to RAI 05.04.06-2-2 in a letter dated July 7, 2009 (ML092190224), states that the RCIC turbine-pump drain leak-off line pump is a nonsafety-related pump, which is used to return leak-off drain water from the turbine drain tank to the RCIC primary pump suction line. In the SER with open items, the staff noted that this RAI would be resolved as part of **Open Item 05.04.06-1,** which is closed as explained above. In response to the audit, Revision 1 to Toshiba Technical Report UTLR-0004-P clarifies that the drain pump performs no safety-related function and is not needed for the TWL pump operation. As the drain pump does not perform a safety-related function, the pump does not need to be included in the IST Program for STP Units 3 and 4. Therefore, **RAI 05.04.06-2-2** is closed.

The shaft bearings in the new RCIC turbine-pump design are in the center between the pump and turbine rotors and are within the single casing. The staff issued **RAI 05.04.06-2-3** requesting the applicant to discuss the IST provisions for the RCIC turbine pump, including vibration monitoring for these bearings. The applicant's response to RAI 05.04.06-2-3 in a letter dated July 7, 2009, states that the IST requirements for the RCIC system are defined in the TS in STP FSAR Tier 2 Chapter 16, as well as Part 4 of the STP COL application. IST activities for RCIC components will satisfy the IST Program developed for STP Units 3 and 4 in accordance with 10 CFR 50.55a. The staff's review of the IST Program for safety-related pumps and valves in the RCIC system for STP Units 3 and 4 is discussed in Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," of this SER. Therefore, **RAI 05.04.06-2-3** is closed.

The control of the new RCIC turbine pump for STP Units 3 and 4 is internal to the pump and turbine with fewer control components than in the previous design. The staff issued **RAI 05.04.06-2-4** requesting the applicant to describe the qualifications and periodic testing of the RCIC turbine-pump control system. The applicant's response to **RAI 05.04.06-2-4** dated July 7, 2009, states that the control system is qualified as part of the overall qualification of the RCIC pump turbine by the pump supplier. For example, the qualification includes verification of the performance of the control system in a pump-turbine assembly while mounted on a shaker table that simulates seismic events. The qualification of the non-metallic components is performed by an analysis of reference data applicable to the materials being used in the control system. As specified in Toshiba Technical Report UTLR-0004-P, the RCIC turbine pump must satisfy the environmental and seismic qualification requirements in ABWR DCD Tier 2 Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," and Section 3.11, "Environmental Qualification of Safety-Related Mechanical and Electrical Equipment"; as well as the environmental and seismic qualification requirements identified in IEEE Std 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE Std 344, "Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations." Proper control system operation will be verified as part of the periodic testing defined in the TS. ABWR DCD Tier 2 Sections 3.9, 3.10, and 3.11 reference the application of IEEE Std 323 and Std 344 for the ABWR design as accepted for use in RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," and RG 1.100 with specific staff positions. As noted above, Revision 1 to Toshiba Technical Report UTLR-0004-P specifies that the TWL pump will be functionally qualified to perform its required functions in compliance with ASME Standard QME-1-2007, as accepted in Revision 3 to RG 1.100. The NRC staff finds the qualification and testing requirements for the RCIC turbine-pump control system acceptable as specified in the ABWR DCD, including ASME Standard QME-1–2007 as discussed in Revision 3 to RG 1.100, and the applicable IEEE standards. Therefore, **RAI 05.04.06-2-4** is closed.

The new RCIC turbine-pump design provides for the main process water to lubricate the bearings in the RCIC turbine pump. The staff issued **RAI 05.04.06-2-5** requesting the applicant to discuss the qualifications of the bearings for water coolant. The applicant's response to

RAI 05.04.06-2-5 in a letter dated July 7, 2009, states that vendor certification includes specifying that the RCIC turbine pump will operate to the design specifications based on water quality requirements for the design. The bearings will be made from carbon materials that are specifically designed to be used with process water lubrication. The qualification will be accomplished by a combination of vendor testing and analysis. As noted above, Revision 1 to Toshiba Technical Report UTLR-0004-P specifies that the TWL pump will be functionally qualified to perform its required functions in compliance with ASME Standard QME-1-2007, as accepted in Revision 3 to RG 1.100. In particular, ASME Standard QME-1-2007 includes qualification specifications that will apply acceptance criteria for the qualification of the RCIC turbine-pump bearings, as well as the filter for the process fluid lubricating the bearings, with respect to functionality and allowable process fluid conditions (such as temperature, impurities, and debris). The NRC staff finds that the functional qualification of the RCIC turbine pump, in accordance with ASME Standard QME-1–2007 and with the staff positions in Revision 3 to RG 1.100, will address the qualification of the RCIC turbine-pump bearings the associated process fluid filter. The functional qualification is therefore acceptable, and **RAI 05.04.06-2-5** is closed.

The staff issued **RAI 05.04.06-2-6** requesting the applicant to describe the qualifications of the RCIC turbine pump and its related components, including stop and throttle valves, for performance and dynamic and seismic conditions. The staff also requested the applicant to describe the environmental qualification process for the electrical and mechanical components, including non-metallic components, for the RCIC turbine pump and its related components. The applicant's response to **RAI 05.04.06-2-6** dated July 7, 2009, states that the qualification of the RCIC turbine pump is consistent with the provisions in Section QR, "General Requirements," and Section QP, "Qualification of Active Pump Assemblies," of ASME Standard QME-1–2007, with the exception that there is no shaft/seal system in the new RCIC turbine-pump design. The TWL pump design is also subject to the seismic and environmental qualification requirements specified in ABWR DCD Tier 2 Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," and Section 3.11, "Environmental Qualification of Safety-Related Mechanical and Electrical Equipment," respectively. The vendor qualification for performance and dynamic and seismic conditions is performed through a combination of factory testing and engineering analysis, in accordance with the applicable ASME Code. For example, the TWL pump design will be demonstrated to satisfy the requirements of IEEE Std. 323 for the environmental conditions inside the reactor building, for normal and accident conditions, and for the seismic requirements of IEEE Std. 344. Revision 1 to Toshiba Technical Report UTLR-0004-P specifies that the TWL pump will be qualified in accordance with the environmental and seismic qualification provisions in ABWR Tier 2 DCD Sections 3.10 and 3.11, IEEE Std. 323 and Std 344, and ASME Standard QME-1–2007 as accepted in Revision 3 to RG 1.100. The NRC staff finds the qualification of the RCIC turbine pump acceptable in accordance with the ABWR DCD provisions, including ASME Standard QME-1-2007 as addressed in Revision 3 to RG 1.100, and the applicable IEEE standards. Therefore, **RAI 05.04.06-2-6** is closed.

In **RAI 05.04.06-2-7**, the NRC staff requested the applicant to discuss the implementation of the maintenance requirements in 10 CFR 50.65 for the RCIC turbine pump and its related components. The applicant's response to **RAI 05.04.06-2-7** dated July 7, 2009, references Section 17.6S, "Maintenance Rule Program," in the STP FSAR. The applicant states that detailed procedures for compliance with 10 CFR 50.65 will be identified before fuel loading through documented instructions and drawings that will include the RCIC System. Toshiba Technical Report UTLR-0004-P provides examples of preventive maintenance for the new RCIC turbine pump, including recommended monthly and annual checks and tests as well as a 5-year maintenance plan. The staff found the STP specifications for maintenance of the RCIC system to be acceptable for this review of the new RCIC turbine-pump design. The overall NRC review of the Maintenance Rule Program is discussed in Section 17 of this SER. The actions by the applicant to develop and implement the Maintenance Rule Program will be within the scope of future inspection activities following COL issuance. Therefore, RAI **05.04.06-2-7** is closed.

In **RAI 05.04.06-2-8,** the staff requested the applicant to describe the quality assurance (QA) provisions for the design and manufacture of the RCIC turbine pump and its related components. The applicant's response to **RAI 05.04.06-2-8** dated July 7, 2009, states that the RCIC pump/turbine vendor QA Program applies the requirements specified in 10 CFR Part 50, Appendix B. ABWR DCD Tier 2, Section 5.4.6, indicates that the RCIC system is part of the Emergency Core Cooling System, which is a safety-related system within the scope of the QA requirements of 10 CFR Part 50, Appendix B. Further, the RCIC turbine pump will be designed, manufactured, and documented in accordance with ASME Code Section III, Division 1, Class 2 requirements. The NRC regulations in 10 CFR 50.55a incorporate by reference the provisions in Section III of the ASME BPV Code, including the applicable QA requirements. The NRC staff finds the application of the QA requirements specified in 10 CFR Part 50, Appendix B, and the ASME BPV Code requirements for the design and manufacture of the RCIC turbine pump to be acceptable. Therefore, **RAI 05.04.06-2-8** is closed.

In **RAI 05.04.06-2-9**, the staff requested the applicant to describe the valves used in the control system for the RCIC turbine pump design and their design, qualification, and IST provisions. The applicant's response to RAI 05.04.06-2-9 dated July 7, 2009, describes the RCIC turbine pump control system and indicates that the qualification process and IST provisions for the RCIC system will include the valves used in the control system. As noted above, Revision 1 to Toshiba Technical Report UTLR-0004-P specifies that the TWL pump will be functionally qualified to perform its required functions, in compliance with ASME Standard QME-1-2007 as accepted in Revision 3 to RG 1.100. As discussed in Section 3.9.6 of this SER, IST activities for the RCIC turbine pump will be provided as specified by the IST Program for STP Units 3 and 4. NRC staff finds the provisions in the ABWR DCD including ASME Standard QME-1-2007 as discussed in Revision 3 to RG 1.100, the applicable IEEE standards, and the IST provisions discussed in Section 3.9.6 of this SER to be acceptable for the design, qualification, and IST activities for the valves in the RCIC turbine pump control system. Therefore, **RAI 05.04.06-2-9** is closed.

In **RAI 05.04.06-2-10**, the staff requested the applicant to discuss the design, qualification, and IST provisions for the steam stop valve and throttle valve for the RCIC turbine-pump design. The applicant's response to RAI **05.04.06-2-10** dated July 7, 2009, states that the qualification, maintenance, and inservice testing for the steam stop valves and throttle valves are included as part of the activities for the RCIC system. As noted above, Revision 1 to Toshiba Technical Report UTLR-0004-P specifies that the TWL pump will be functionally qualified to perform its required functions in compliance with ASME Standard QME-1–2007 as accepted in Revision 3 to RG 1.100. As discussed in Section 3.9.6 of this SER, IST activities for the RCIC turbine pump will be provided as specified by the IST Program for STP Units 3 and 4. The NRC staff finds the provisions in the ABWR DCD, including ASME Standard QME-1–2007 as discussed in Revision 3 to RG 1.100, the applicable IEEE standards, and the IST provisions discussed in Section 3.9.6 of this SER to be acceptable for the design, qualification, and IST activities for the steam stop valve and throttle valve for the RCIC turbine pump design. Therefore, **RAI 05.04.06-2-10** is closed.

In **RAI 05.04.06-2-11**, the staff requested the applicant to discuss the design, qualification, and IST provisions for a solenoid-operated, four-way crossover valve used in the RCIC turbine pump. The applicant's response to RAI **05.04.06-2-11** dated July 7, 2009, states that the qualification, maintenance, and inservice testing for this valve will be included as part of those activities for the RCIC system. As noted above, Revision 1 to Toshiba Technical Report UTLR-0004-P specifies that the TWL pump will be functionally qualified to perform its required functions in compliance with ASME Standard QME-1–2007, as accepted in Revision 3 to RG 1.100. As discussed in Section 3.9.6 of this SER, IST activities for the RCIC turbine pump will be provided as specified by the IST Program for STP Units 3 and 4. The staff finds the provisions in the ABWR DCD, including ASME Standard QME-1–2007 as discussed in Revision 3 to RG 1.100, the applicable IEEE standards, and the IST provisions discussed in Section 3.9.6 of this SER to be acceptable for the design, qualification, and IST activities for the crossover valve for the RCIC turbine pump design. Therefore, **RAI 05.04.06-2-11** is closed.

ABWR DCD Tier 2 Subsection 3.9.2.1.1, "Vibration and Dynamic Effects Testing," describes tests to confirm that piping, components, restraints, and supports have been designed to withstand the dynamic effects of steady-state, flow-induced vibration (FIV) and anticipated operational transient conditions. In **RAI 05.04.06-2-12**, the staff requested the applicant to summarize the consideration of potential adverse flow effects from severe hydrodynamic and acoustic resonance loads on the RCIC system and its components. The applicant's response to RAI **05.04.06-2-12** dated July 7, 2009, states that the change in the RCIC pump design does not affect the key design parameters that might impact hydrodynamic and acoustic resonance loads from flow effects. The new RCIC turbine pump will be qualified to withstand the vibratory loads associated with a combination of seismic and hydrodynamic loads that are greater than those experienced by the ABWR. Monitoring of the piping movement in response to loads from flow effects will be performed as part of the Startup Test Program for STP Units 3 and 4, as described in ABWR DCD Tier 2 Subsection 14.2.12.2.11, "System Vibration," and Subsection 14.2.12.2.22, "RCIC System Performance," incorporated by reference and modified in the STP FSAR. The applicant's response to RAI 03.09.06-26 dated August 17, 2009 (ML092310488) states that the IST Program will address the dynamic effects of steady-state FIV and anticipated transient conditions as they relate to pumps, valves, and dynamic restraints. The applicant also states that a parallel program that takes advantage of nuclear power plant operating experience will be developed to assure that hydrodynamic loads and acoustic resonance are considered in the design of the reactor coolant, steam, and feedwater systems. The staff considered the provisions in ABWR DCD Tier 2, Subsection 3.9.2.1.1, and DCD Tier 2 Chapter 14, with the clarification in the RAI response to provide confidence that the impacts of dynamic effects from FIV on the RCIC system will be addressed at STP Units 3 and 4. The actions of the applicant to address potential FIV effects will be within the scope of future inspection activities following COL issuance. **RAI 05.04.06-2-12** is therefore closed.

In **RAI 05.04.06-2-13**, the staff requested the applicant to describe the analysis requirements, system specifications, and pump performance curves for the RCIC turbine pump to demonstrate that the new design will satisfy the ABWR requirements. The applicant's response to **RAI 05.04.06-2-13** dated July 7, 2009, references Toshiba Technical Report UTLR-0004-P for the requested information. The staff found the RCIC technical report acceptable for describing the analysis requirements, system specifications, and pump performance curves. Therefore, **RAI 05.04.06-2-13** is closed.

COL License Information Item

- COL License Information Item 5.8 Analyses of 8-hour RCIC Capability

Analysis to Demonstrate the Facility Has an 8–Hour, Non-Design SBO Capability

The applicant states that the best estimate analysis will be available for NRC staff to review by the end of preoperational testing demonstrating that the RCIC system can function for 8 hours in an SBO event. The applicant further states that the analysis will reflect Class 1E loadings based on expected plant and operator responses during this event. The applicant states that "Similar evaluations have been satisfactorily performed for other plants (COM 5.4-1)." Because a best estimate analysis will be available for NRC to review by the end of preoperational testing demonstrating that the RCIC system can function for 8 hours in an SBO event, this COL license information item is satisfied. This satisfies COL License Information Item No. 5.8, which is identified in DCD Table 1.9-1.

Analysis to Demonstrate that the DC Batteries and SRVs/Automatic Depressurization System (ADS) Pneumatics Have Sufficient Capacity

A best estimate analysis demonstrating that the DC batteries and SRVs/ADS pneumatics have sufficient capacity to open and maintain the necessary safety relief valves open to depressurize the RCS following a RCIC failure due to battery failure (at about 8 hours), so that the AC-independent water addition mode of the RHR system can inject to the core (COM 5.4-2). The applicant states that a best estimate analysis demonstrating adequate DC battery and pneumatic supply capacity based on the as-purchased equipment configuration will be completed and available for NRC review before the commencement of the Preoperational Test Program. This response satisfies the COL License Information Item No. 5.8 identified in DCD Table 1.9-1.

5.4.6.5 *Post Combined License Activities*

The applicant identifies the following commitments:

- Commitment (COM 5.4-1) – Demonstrate that the facility has the 8-hour non-design basis SBO capability.

- Commitment (COM 5.4-2) – Demonstrate that DC batteries and SRV/ADS pneumatics have sufficient capacity to open and maintain open SRVs that are necessary to depressurize the reactor coolant system following an RCIC failure due to battery failure (at about 8 hours).

5.4.6.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to the RCIC system, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the RCIC system that were incorporated by reference have been resolved.

The staff concludes that STP FSAR Section 5.4.6 adequately incorporates by reference ABWR DCD Tier 2, Section 5.4.6. The staff's review confirmed that the applicant has adequately supported the supplements to and departures from the ABWR DCD addressed in STP FSAR Section 5.4.6. The staff determined that the applicant is in compliance with the NRC regulations.

5.4.7 Residual Heat Removal System

5.4.7.1 *Introduction*

This section of the FSAR addresses the RHR system in several different operational configurations, including the four primary functions:

a. The RHR system is used to cool the RCS during and following shutdown. For the RCS cooldown, RHR is used in conjunction with the feedwater system and the main condenser.

b. Parts of the RHR system also act to provide low-pressure emergency core cooling and compensation for reactor vessel water inventory loss. The low-pressure emergency core cooling function is reviewed in Section 6.3 of the application.

c. In the wetwell and drywell spray cooling mode, the RHR system provides containment heat removal capability. The containment heat removal function is reviewed in Section 6.2.2 of the application.

d. The RHR provides suppression pool (S/P) cooling. In addition, secondary functions of the RHR include fuel pool cooling (FPC), pool draining, and AC-independent water addition (ACIWA).

5.4.7.2 *Summary of Application*

Section 5.4.7 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.7 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in COL FSAR Section 5.4.7, the applicant provides the following:

<u>*Tier 1 Departure*</u>

- STD DEP T1 2.4-1 — Residual Heat Removal System and Spent Fuel Pool Cooling

This departure describes a design change that adds the capability to allow the choice of a third loop, RHR division A, in the augmented FPC and fuel pool makeup modes.

<u>*Tier 2 Departures Not Requiring Prior NRC Approval*</u>

- STD DEP 5.4-3 — Residual Heat Removal System Interlock

This departure clarifies that the minimum flow valves open logic in Table 5.4-3 is consistent with the figures.

- STD DEP 6C-1 Containment Debris Protection for ECCS Strainers

This departure incorporates the new, complex design of the ECCS strainers (e.g., cassette-type strainer) per NUREG/CR-6224, NUREG/CR–6808, and guidance from "Utility Resolution Guidance for ECCS Strainer Blockage," NEDO–32868-A.

- STD DEP Vendor

This departure describes the applicant's decision to use the services of an alternate vendor to support the application.

- STD DEP 5B-1 Residual Heat Removal Flow and Heat Capacity Analysis

This departure increases the heat removal capacity of the RHR heat exchanges to allow for a reduced outage time.

COL License Information Item

- COL License Information Item 5.9 ACIWA Flow Reduction

This COL license information item addresses the hydraulic analysis that will be performed to determine if a flow reduction device is needed based on the actual flow rate capacities, pressure, and hose size of the diesel-driven pump. This analysis will be available for NRC review before the commencement of the Preoperational Test Program. (COM 5.4-3).

5.4.7.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the RHR system, and the associated acceptance criteria, are in Section 5.4.7 of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 1 and Tier 2 departures. Tier 1 departures requiring prior NRC approval are subject to the requirements specified in 10 CFR Part 52 Appendix A, Section VIII.A.4. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

The regulatory basis for reviewing the COL license information items is in Section 5.4.7 of NUREG–0800.

5.4.7.4 *Technical Evaluation*

As documented in NUREG–1503, the staff reviewed and approved Section 5.4.7 of the certified ABWR DCD. The staff reviewed Section 5.4.7 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the RHR system.

The staff reviewed the applicant's proposal using the review procedures described in Section 5.4.7 of NUREG–0800, and Branch Technical Position (BTP) 5-4. The staff's review of this application includes the following considerations:

Tier 1 Departure

- STD DEP T1 2.4-1 Residual Heat Removal System and Spent Fuel Pool Cooling

In FSAR Subsection 5.4.7.1, "Design Basis," the applicant introduces STD DEP T1 2.4-1, which revises the number of RHR loops connected to the upper pools from two to three to provide additional flexibility in shutdown cooling flow to the upper pools, during normal refueling activities. The purpose of this departure is to improve the capability of performing divisional outages in any order for maintenance and other activities, while maintaining the single failure margin. Therefore, the change will add RHR division A loop in the augmented FPC and fuel pool makeup modes in addition to divisions B and C. The applicant states that the additional components, such as piping and valves, "will be of the same quality standard, seismic category, and ASME code as the B and C RHR loop components." The applicant also incorporated STD DEP T1 2.4-1 in FSAR Subsections 5.4.7.1.1.8, "Fuel Pool Cooling"; 5.4.7.2.6, "Manual Action"; and Figure 5.4-11, "RHR Process Flow Diagram (PFD) (sheets 1 & 2)."

In Tier 1 Section 2.4.1, the RHR is also revised to reflect the new Division A connection to the FPC System. The cross-tie connections are also correctly shown in the Tier-1 revisions of Figures 2.4.1a, b, and c. ITAAC Table 2.4.1, item # 7 is revised to include the Division A connection. NRC staff concluded that the proposed change, which includes additional components and operational procedural changes, will not result in a decrease in the level of safety. Therefore, the staff found Departure STD DEP T1 2.4-1 acceptable.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 5.4-3 Residual Heat Removal System Interlock

The applicant incorporates the STD DEP 5.4-3 Departure, which is composed of three individual revisions related to RHR system interlock logic, into several parts of FSAR Section 5.4. In FSAR Subsection 5.4.7.1.1.6, "Wetwell and Drywell Spray Cooling," the applicant notes a discrepancy between this section and Figure 7.3-4 sheet 11 of 20, in respect to the mode of operation for initiation of only the wetwell spray function. Because the applicant has determined that Figure 7.3-4 is in conformance with the ABWR design, the words "one of the full flow modes, which are either" and "or the low pressure flooder (LPFL) mode" were removed from the fifth sentence of this section to state, "If wetwell spray is desired by itself, without drywell spray, it can be initiated by operator action, but must be used in conjunction with the suppression pool (S/P) cooling mode." In conjunction with Subsection 5.4.7.1.1.6, the staff reviewed sheet 11 of Figure 7.3-4 and the logic sequencing of the various RHR modes in FSAR Subsection 7.3.1.1,

1 See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

‘Systems Descriptions,” and it was noted that the wetwell spray mode logic sequencing does not mention or reference that the wetwell spray mode “must be used in conjunction with the suppression pool (S/P) cooling mode” to support the description of Subsection 5.4.7.1.1.6. NRC staff issued **RAI 05.04.07-1** requesting an explanation of the S/P cooling mode with respect to the wetwell spray mode.

The applicant’s response dated June 1, 2009 (ML091540277), provides additional information that adequately addresses **RAI 05.04.07-1**. The applicant states, “Initiation of suppression pool cooling in conjunction with wetwell cooling is a manual operation and is not a result of any instrumentation or control logic.” Therefore, the manual initiation of the S/P cooling mode is not applicable to the instrumentation or control logic described in Subsection 7.3.1.1.3. The staff found the applicant’s response acceptable and **RAI 05.04.07-1** is resolved. Also, the staff noted several editorial errors in the wetwell logic sequencing subsection and issued **RAI 05.04.07-2**. The applicant’s response to this RAI dated June 1, 2009, agrees to correct the errors noted in the RAI in the next revision of the COL. Therefore, the staff found the response acceptable and RAI 05.04.07-2 was tracked as **Confirmatory Item 05.04.07-2.** The staff confirmed in revision 4 of the FSAR that the proposed changes were included in the COL application. Therefore RAI 05.04.07-2 and Confirmatory Item 05.04.07-2 are closed.

The second change defined in the STD DEP 5.4-3 Departure refers to the revision of DCD Tier 2 Table 5.4-3, Note C. Note C was revised from “Pump is running” to “Pump discharge pressure high and low loop flow signal” to conform with DCD Tier 2, Figure 7.3-4 sheet 12, which is in accordance with the ABWR design. From sheet 12 of Figure 7.3-4, the staff confirmed that the revision was for clarification and consistency, with no impact on the ABWR design. Therefore, the staff found this revision acceptable.

The final revision described in the STD DEP 5.4-3 Departure relates to the change to relief pressure values of E11-F028A-C and E11- F051A-C in DCD Tier 2, Table 5.4-5. The relief pressure values were changed from 3.44 MPaG to 3.43 MPaG to be consistent with DCD Tier 2 Figure 5.4-10 sheets 3, 4, and 6. The staff confirmed the values in Figure 5.4.3-10 and concluded that the change is conservative. However, because the change represents a difference of approximately 1.5 psi, the staff expressed concerned as to whether the revised values affect the outcome of any analysis, and issued **RAI 05.04.07-3.**

In the response to this RAI dated June 1, 2009 (ML091540277), the applicant explains that the change from 3.44 MPaG to 3.43 MPaG is intended to correct a typographical error in DCD Tier 2, Table 5.4-5. In DCD Figure 5.4-10 sheets 3, 4, and 6, the correct setpoint of 3.43 MPaG is displayed. In addition, the calculation of the peak reactor pressure is not affected by this change because the analysis is based on the correct setpoint value of 3.43 MPaG. Therefore, the staff concluded that the applicant’s response is acceptable and **RAI 05.04.07-3** is resolved.

- STD DEP 6C-1 Containment Debris Protection for ECCS Strainers

This departure incorporates the new complex ECCS strainers such as the cassette-type strainer design per NUREG/CR–6224, NUREG/CR–-6808, and NEDO–32868-A. The new strainer design affects the available NPSH of the high-pressure core flooder (HPCF) and low-pressure core flooder (LPCF) RHR pumps.

In FSAR Subsection 5.4.7.2.2, “Equipment and Component Description,” the applicant indentifies STD DEP 6C-1, which replaces the present stacked disk ECCS strainer design with a new complex ECCS strainer design having a larger surface area that reduces the risk of blockage. Because strainers are passive components, the applicant states that the new strainer does not impact the operation of the ABWR or the frequency of occurrence of an accident but improves the reliability of the RHR. However, NRC staff noted that the NPSH was changed from 2.4 m to 2.0 m, which equates to an approximate 16 percent reduction in the NPSH, and issued **RAI 05.04.07-4.** The applicant's response to this RAI dated June 1, 2009, notes that the NPSH reduction applies to the required NPSH and not to the available NPSH for the RHR pump. The staff concluded that the applicant's response adequately clarifies the concerns of this RAI and is thus acceptable. Therefore**, RAI 05.04.07-4** is resolved.

In a letter dated October 29, 2009 (ML093090336), the applicant submitted a supplemental response to **RAI 06.02.02-6** of the RHR and HPCF pump calculations in regard to the available NPSH margin. In Table 4-5 of Reference 3 of this supplemental submittal, the calculated available NPSH for the ECCS pumps is 2.17 meters (7.12 feet). In addition, the calculated ECCS pump required NPSH is 2.0 meters (6.56 feet); therefore, adequate NPSH margin is available for the ECCS pumps to perform their functions as described in the design basis.

- STD DEP 5B-1 Residual Heat Removal Flow and Heat Capacity Analysis

This Departure is evaluated in Chapter 6 of this SER.

- STD DEP Vendor

This departure removes all references to GE from the FSAR text. The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval.

<u>*COL License Information Item*</u>

- COL License Information Item 5.9 ACIWA Flow Reduction

Prior to the commencement of the Preoperational Test Program, the applicant committed (COM 5.4-3) to perform a hydraulic analysis to determine the need for a flow reduction device. The analysis will be based on the actual flow rate capacities, pressure, and hose size of the diesel-driven pump. NRC staff concluded that COL License Information Item 5.9 is acceptable.

5.4.7.5 *Post Combined License Activities*

The applicant identifies the following commitment:

- Commitment (COM 5.4-3) – Perform a hydraulic analysis to determine whether a flow-reduction device is needed before the commencement of the Preoperational Test Program.

5.4.7.6 *Conclusion*

NRC staff's finding related to information incorporated by reference is in NUREG–1503. The staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information related to RHR system. No outstanding information is expected to be addressed in the COL FSAR related to this section.[1] Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the RHR system that were incorporated by reference have been resolved.

The staff's review concluded that the applicant's proposed resolution to COL License Information Item 5.6 meets NRC regulations and the relevant acceptance criteria of SRP Section 5.4.7.

5.4.8 Reactor Water Cleanup System

5.4.8.1 *Introduction*

This section of the SER addresses the reactor water cleanup system (RWCS).

5.4.8.2 *Summary of Application*

Section 5.4.8 of the STP COL FSAR incorporates by reference Section 5.4.8 of the ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 5.4.8, the applicant provides the following:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 5.4-1 Reactor Water Cleanup system

This departure describes changes to the RWCS design; specifically to the cleanup system pumps and doubling their capacity.

5.4.8.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503.

In addition, the relevant requirements of the Commission regulations for the reactor water cleanup system, and the associated acceptance criteria, are in Section 5.4.8 of NUREG–0800.

In accordance with Section VIII, "Processes for Changes and Departures," of "Appendix A to Part 52--Design Certification Rule for the U.S. Advanced Boiling Water Reactor," the applicant identifies Tier 2 departures. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

In addition, the regulatory basis for the RWCS consideration is in the following:

- GDC 1, as it relates to the design of the RWCS and components to standards commensurate with the importance of its safety function.

- GDC 2, as it relates to the ability of the RWCS to withstand the effects of natural phenomena.

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1 See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

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

- GDC 14, as it relates to ensuring the integrity of the reactor coolant pressure boundary.

- GDC 15, as it relates to reactor coolant-associated auxiliary system design with a sufficient margin.

- GDC 31, as it relates to fracture prevention or the RCPB design with a sufficient margin.

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

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

5.4.8.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 5.4.8 of the certified ABWR DCD. The staff reviewed Section 5.4.8 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the RWCS.

The staff's review of this application includes the following considerations:

<u>*Tier 2 Departure Not Requiring Prior NRC Approval*</u>

- STD DEP 5.4-1 Reactor Water Cleanup System

NRC staff's review found that this departure does not affect the technical evaluation of the RWCS documented in Section 5.4.8 of the ABWR FSER.

The applicant's evaluation determined that this departure does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5. Within the review scope of this section, the staff finds it reasonable that the departure does not require prior NRC approval. The applicant's process for evaluating departures and other changes to the certified ABWR DCD is subject to NRC inspections.

5.4.8.5 *Post Combined License Activities*

There are no post COL activities related to this section.

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1 See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

5.4.8.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the RWCS that were incorporated by reference have been resolved.

The staff found it reasonable that the identified Tier 2 departures are characterized as not requiring prior NRC approval, per 10 CFR Part 52, Appendix A, Section VIII.B.5.

5.4.9 Main Steamlines Feedwater Piping

Section 5.4.9 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.9, "Main Steamlines Feedwater Piping," of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the main steamlines feedwater piping have been resolved.

5.4.10 Pressurizer

This section is not applicable to the ABWR.

5.4.11 Pressurizer Relief Valves

This section is not applicable to the ABWR.

5.4.12 Valves

Section 5.4.12 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.12, "Valves," of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the valves have been resolved.

5.4.13 Safety/Relief Valves

Section 5.4.13 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.13, "Safety/Relief Valves," of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding issue related to this section. Pursuant

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[1] See *"Finality of Referenced NRC Approvals"* in SER Section 1.1.3, for a discussion on the staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

to ~~10~~ 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the safety/relief valves have been resolved.

5.4.14 Component Support

Section 5.4.14 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 5.4.14, “Component Supports,” of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff’s review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the component supports have been resolved.