15.0 TRANSIENT AND ACCIDENT ANALYSES

15.0.1 Introduction

This section of the South Texas Project (STP), Units 3 and 4, Combined License (COL) Final Safety Analysis Report (FSAR) addresses the evaluation of the safety of a nuclear power plant and includes analyses of the plant's responses to postulated disturbances in process variables and postulated equipment failures or malfunctions. The safety analyses provide a significant contribution to the selection of limiting conditions for plant operation, limiting safety system settings, and design specifications for plant components and systems from the standpoint of public health and safety.

15.0.2 Summary of Application

Section 15.0, "Accident and Analysis," of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Section 15.0 of the certified 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, "Licenses, Certifications, And Approvals For Nuclear Power Plants," Appendix A "Design Certification Rule for the U.S. Advanced Boiling Water Reactor," with no departures or supplements. In addition, in FSAR Section 15.0, the applicant provided the following:

<u>COL License Information Items</u>

- COL License Information Item 15.1 Anticipated Operational Occurrences

This COL license information item addresses the Anticipated Operational Occurrences (AOOs).

- COL License Information Item 15.2 Operating Limits

This COL license information item addresses plant operating limits.

- COL License Information Item 15.3 Design-Basis Accidents

This COL license information item addresses design-basis accidents (DBAs).

15.0.3 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) [Final Safety Evaluation Report (FSER) related to the certified ABWR DCD.]

In addition, the relevant requirements of the U.S. Nuclear Regulatory Commission (NRC) regulations for the transient and accident analyses, and the associated acceptance criteria, are in Chapter 15 of NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, (LWR Edition)," the Standard Review Plan (SRP).

The acceptance criteria for reviewing the COL license information items are in Section 15 of NUREG–0800.

15.0.4 Technical Evaluation

As documented in NUREG–1503, the NRC staff reviewed and approved Section 15.0 of the certified ABWR DCD. The NRC staff reviewed Section 15.0 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 NRC 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 NRC staff reviewed the following information in the COL FSAR:

COL License Information Items

- COL License Information Item 15.1 Anticipated Operational Occurrences

In Revision 3 of FSAR Subsection 15.0.5.1, the applicant stated, "The analysis results of the events identified in Subsection 15.0.4.5 for initial core loading will be prepared and provided as an amendment to the FSAR in accordance with 10 CFR 50.71 (e) at least one year prior to fuel load. This analysis will reflect the final fuel design for the initial core. (COM 15.0-1)"

- COL License Information Item 15.2 Operating Limits

In Revision 3 of FSAR Subsection 15.0.5.2, the applicant stated, "The operating limit resulting from the analyses normally provided in this subsection will be prepared and provided as an amendment to the FSAR in accordance with 10 CFR 50.71 (e) at least one year prior to fuel load. This analysis will reflect the final fuel design for the initial core. (COM 15.0-2)"

- COL License Information Item 15.3 Design-Basis Accidents

In Revision 3 of FSAR Subsection 15.0.5.3, the applicant stated, "The results of the DBAs associated with the initial core, including radiological consequences, will be prepared and provided as an amendment to the FSAR in accordance with 10 CFR 50.71 (e) at least one year prior to fuel load. This analysis will reflect the final fuel design for the initial core. (COM 15.0-3)."

For the items listed below, the applicant has proposed to provide the required information as an amendment to the FSAR, at least one year prior to fuel load:

- FSAR Section 15.0.5.1, COL Information Item 15.1 (COM 15.0-1)
- FSAR Section 15.0.5.2, COL Information Item 15.2 (COM 15.0-2)
- FSAR Section 15.0.5.3, COL Information Item 15.3 (COM 15.0-3)

In request for additional information (RAI) 04.04-2, the NRC staff informed the applicant that the method proposed by the applicant, is not an acceptable resolution. In its response to RAI

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

04.04-2, dated November 19, 2009 (ML093270045), the applicant stated that no departures are taken from the fuel design licensing basis that is described in the ABWR DCD, including the core loading map used for the transient and accident response analysis in DCD Figure 4.3-1, "Core Loading Map used for Response Analyses," and the control rod strategy in DCD Table 4A-1, "Basic Control Strategy for Typical ABWR." Because the certified DCD Chapter 15 includes the necessary analysis required for the core being licensed, COL License Information Items 15.1, 15.2, and 15.3 are considered closed and Commitment (COM 15.0-1) through Commitment (COM 15.0-3) are closed. Therefore, the applicant has withdrawn these commitments.

15.0.5 Post Combined License Activities

There are no post COL activities related to this section.

15.0.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. The NRC staff reviewed the application and checked the referenced DCD. The NRC 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 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the transient and accident analyses that were incorporated by reference have been resolved.

In addition, the NRC staff compared the additional information in the COL application (COLA) to the relevant NRC regulations, the guidance in Chapter 15 of NUREG–0800, and other NRC regulatory guides (RGs). The NRC staff's review concluded that the applicant has adequately addressed COL License Information Items 15.1 through 15.3 in accordance with Chapter 15 of NUREG-0800.

15.1 Decrease in Reactor Coolant Temperature

15.1.1 Introduction

This section of the FSAR addresses the AOOs that increase heat removal by the steam and feedwater systems causing a decrease in reactor coolant temperature. Increased heat removal can be caused by:

- Loss of feedwater heating.
- Feedwater controller failure.
- Pressure regulator failure (maximum demand).
- Inadvertent safety/relief valve opening.
- Inadvertent residual heat removal shutdown cooling operation.

15.1.2 Summary of Application

Section 15.1, "Decrease in Reactor Coolant Temperature," of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Section 15.1 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 15.1, the applicant provided the following:

Supplemental Information

In FSAR Revision 3, Subsection 15.1.2.3.2.2, the applicant commits (COM 15.1-1) to provide an analysis of a feedwater controller failure maximum-demand transient reflecting the final fuel design for the initial core loading.

In FSAR Revision 2, Table 15.1S-2, "Instrument Response Time," the applicant provided instrument response times for STP, Units 3 and 4.

15.1.3 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 decrease in reactor coolant temperature, and the associated acceptance criteria, are in Section 15.1.1 - 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve," of NUREG–0800.

15.1.4 Technical Evaluation

As documented in NUREG–1503, the NRC staff reviewed and approved Section 15.1 of the certified ABWR DCD. The NRC staff reviewed Section 15.1 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.[1] The NRC 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 NRC staff reviewed the following information in the COL FSAR:

Supplemental Information

In its response to a COL license information item in ABWR DCD Subsection 15.1.2.3.2.2, the applicant provided the following site-specific Commitment (COM 15.1-1) as supplemental information in FSAR Revision 3, Subsection 15.1.2.3.2.2, "Feedwater Controller Failure-Maximum Demand:"

> The analysis for the initial core will be prepared and provided as an amendment to the FSAR in accordance with 10 CFR 50.71 (e), at least one year prior to the fuel load. This analysis will reflect the final fuel design for the initial core loading. (COM 15.1-1).

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

In RAI 04.04-2, the NRC staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. In its response to RAI 04.04-2, dated November 19, 2009 (ML093270045), the applicant stated that no departures are taken from the fuel design licensing basis that is described in the ABWR DCD, including the core loading map used for the transient and accident response analysis in DCD Figure 4.3-1 and the control rod strategy in DCD Table 4A-1. Because certified DCD Subsection 15.1.2.3.2.2 includes the feedwater controller failure maximum-demand analysis required for the core being licensed, the applicant's Commitment (COM 15.1-1) is considered closed. Therefore, the applicant has withdrawn this commitment.

The applicant submitted a new comparison —Table 15.1S-2—listing the instrument response times stated in the DCD for STP, Units 3 and 4. There was a significant change in the response time from the DCD values assumed in the analysis for the scram, the safety relief function, recirculation pump trip (RPT), and the main steam isolation valve (MSIV). Therefore, the NRC staff issued RAI 15.01.04-1. In its response to RAI 15.01.04-1, dated July 2, 2009 (ML091880283), the applicant stated that the changes were made inadvertently. In FSAR Revision 3, the instrument delay times were returned to the values in the DCD. Because the values were returned to the DCD values, which were determined to be acceptable in NUREG–1503, the NRC staff considers RAI 15.01.04-1 to be resolved and closed.

15.1.5 Post Combined License Activities

There are no post COL activities related to this section.

15.1.6 Conclusion

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

In addition, the NRC staff compared the supplemental information in the COL application to the relevant NRC regulations, the guidance in Section 15.1.1-15.1.4 of NUREG–0800. The NRC staff's review concluded that the applicant has adequately addressed the supplemental information item in accordance with Section 15.1.1-15.1.4 of NUREG–0800.

15.2 <u>Increase in Reactor Pressure</u>

15.2.1 Introduction

This section of the FSAR addresses the AOOs that decrease the heat removal by the steam and feedwater systems causing an increase in reactor pressure. Decreased heat removal can be caused by:

- Pressure regulator failure (closed).
- Generator load rejection.
- Turbine trip.

- MSIV closures.
- Loss of condenser vacuum.
- Loss of alternating current (ac) power to station auxiliaries.
- Loss of feedwater flow.
- Failure of residual heat removal shutdown cooling.

15.2.2 Summary of Application

Section 15.2, "Increase in Reactor Pressure," of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Section 15.2 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 15.2, the applicant provided the following:

Tier 1 Departure

- STD DEP T1 2.3.1 — Deletion of MSIV Closure and Scram on High Radiation

This departure addresses the deletion of the scram and the MSIV automatic closure on the high main steam line radiation monitor trip.

Tier 2 Departure Requiring Prior NRC Approval

- STD DEP 8.3-1 — Plant Medium Voltage Electrical System Design

This departure changes the medium voltage distribution system in the offsite electrical power system, onsite ac power distribution system, and safety loads.

Supplemental Information

In FSAR, Revision 3, Subsection 15.2.1.3.1, the applicant commits (COM 15.2-1) to provide an analysis of the inadvertent closure of one turbine control valve reflecting the final fuel design for the initial core loading.

In FSAR, Revision 3, Subsection 15.2.2.3.2.3, the applicant commits (COM 15.2-2) to provide an analysis of generator load rejection with a failure of all turbine bypass valves reflecting the final fuel design for the initial core loading.

COL License Information Item

- COL License Information Item 15.4 — Radiological Effects of MSIV Closure

This COL license information item addresses the exclusion area boundary (EAB) long-term routine release doses associated with the inadvertent closure of the MSIVs.

15.2.3 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 increase in reactor pressure, and the associated acceptance criteria, are in Section 15.2.1 15.2.5, "Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam pressure Regulator Failure (Closed)," 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 affecting technical specifications (TS) require prior NRC approval and are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.C.4.

15.2.4 Technical Evaluation

As documented in NUREG–1503, the NRC staff reviewed and approved Section 15.2 of the certified ABWR DCD. The NRC staff reviewed Section 15.2 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.[1] The NRC 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 NRC staff reviewed the following information in the COL FSAR:

Tier 1 Departure

The following Tier 1 departure identified by the applicant in this section requires prior NRC approval and the full scope of its technical impact may be evaluated in other sections of this Safety Evaluation Report (SER) accordingly. For more information, refer to the COLA, Part 7, Section 5.0, "Tables and Indexes," for a listing of all FSAR sections affected by this Tier 1 departure.

- STD DEP T1 2.3.1 — Deletion of MSIV Closure and Scram on High Radiation

The applicant's Chapter 15 accident analysis does not take credit for this protective action. The NRC staff's review of the radiological aspects of this departure is in Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems," of this SER.

Tier 2 Departure Requiring Prior NRC Approval

The following Tier 2 departure identified by the applicant in this section, requires prior NRC approval and the full scope of its technical impact may be evaluated in the other sections of this

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

SER accordingly. For more information, refer to the COLA, Part 7, Section 5.0 for a listing of all FSAR sections affected by this Tier 2 departure.

- STD DEP 8.3-1 Plant Medium Voltage Electrical System Design

The NRC staff's review of this departure is in Chapter 8, "Electric Power," Section 8.2, "Offsite Power System," and Section 8.3, "Onsite Power Systems," of this SER.

Supplemental Information

In its response to the COL license information items in ABWR DCD Subsections 15.2.1.3.1 and 15.2.2.3.2.3, the applicant provided the following site-specific Commitments (COM 15.2-1 and 15.2-2) to address the inadvertent closure of one turbine control valve and generator load rejection with the failure of all turbine bypass valves:

> The analysis for the initial core will be prepared and provided as an amendment to the FSAR in accordance with 10 CFR 50.71 (e), at least one year prior to the fuel load. This analysis will reflect the final fuel design for the initial core loading. (COM 15.2-1 and COM 15.2-2)

In RAI 04.04-2, the NRC staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. In its response to RAI 04.04-2, dated November 19, 2009 (ML093270045), the applicant stated that no departures are taken from the fuel design licensing basis that is described in the ABWR DCD, including the core loading map used for the transient and accident response analysis in DCD Figure 4.3-1 and the control rod strategy in DCD Table 4A-1. Because certified DCD Subsections 15.2.1.3.1 and 15.2.2.3.2.3, include the analysis required for the core being licensed, the NRC staff determined that the applicant's commitments adequately address the COL license information items in DCD Subsections 15.2.1.3.1 and 15.2.2.3.2.3 are considered to be closed. Therefore, the staff considers RAI 04.04-2 to be resolved and closed. Therefore, the applicant has withdrawn these commitments.

COL License Information Item

- COL License Information Item 15.4 Radiological Effects of MSIV Closure

This COL license information item addresses the radiological effects of MSIV closures. The information provided, describes the radiological consequences of the inadvertent closure of the MSIV which can be initiated by various steam line and nuclear system malfunctions. The certified ABWR DCD Table 15.2-12, "Dose Evaluation and Meteorology," provides inadvertent MSIV closure doses as a function of atmospheric dispersion factors (χ/Q values).

Subsection 15.2.10.1 of the STP FSAR, Revision 5, stated that the radiological consequences of the inadvertent MSIV closure are 4.5E-4 milliGray (mGy) (4.5E-2 milli-radiation absorbed dose [mrad]) for thyroid dose and 1.3E-2 mGy (1.3 mrad) whole body dose. These dose values are based on: (1) the radiation doses provided in Table 15.2-12 of the ABWR DCD, and (2) the ratio of the referenced χ/Q values in the ABWR DCD to site-specific χ/Q values at the STP site. The radiation doses provided in Table 15.2-12 of the ABWR DCD and the referenced χ/Q values in the ABWR DCD were approved by the NRC staff in its review of the ABWR standard reactor design certification (DC). Therefore, the NRC staff determined the FSAR information on the radiological consequences of the inadvertent MSIV closure to be consistent with the values that were reviewed and approved by the NRC staff in NUREG–1503.

15.2.5 Post Combined License Activities

There are no post COL activities related to this section.

15.2.6 Conclusion

The NRC staff's finding related to information incorporated by reference, is in NUREG-1503. The NRC staff reviewed the application and checked the referenced DCD. The NRC 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 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the increase in reactor pressure that were incorporated by reference have been resolved.

In addition, the NRC staff compared the additional information in the COLA to the relevant NRC regulations, the guidance in Sections 15.2.1 15.2.5 of NUREG–0800. The NRC staff's review concluded that the applicant has adequately addressed COL License Information Item 15.4, the Tier 1 departure, and the supplemental information in accordance with Section 15.2.1 Section 15.2.5 of NUREG–0800, and determined it to be reasonable that the identified Tier 2 departure is characterized as not requiring prior NRC approval per 10 CFR Part 52, Appendix A, Section VIII.B.5.

15.3 <u>Decrease in Reactor Coolant System Flow Rate</u>

This section of the FSAR addresses the AOOs that cause a decrease in the reactor coolant system (RCS) flow rate. A decreased flow rate can be caused by:

- Reactor internal pump trip.
- Recirculation flow controller failure (decreasing flow).
- Pressure regulator downscale failure.

Section 15.3, "Decrease in Reactor Coolant System Flow Rate," of the STP, Units 3 and 4, COL FSAR incorporates by reference Section 15.3, "Decrease in Reactor Coolant System Flow Rate," of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this subsection. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the AOOs that cause a decrease in the reactor RCS flow rate have been resolved.

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

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Introduction

This section of the FSAR, addresses the AOO and the accidents that cause an anomaly in the reactivity or power distribution in the reactor core. Reactivity and power distribution anomalies can be caused by:

- Rod withdrawal errors and malfunctions (low power and at power).
- Mislocated fuel bundle accident.
- Misoriented fuel bundle accident.
- Rod ejection accident.
- Control rod drop accident.

15.4.2 Summary of Application

Section 15.4, "Reactivity and Power Distribution Anomalies," of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Section 15.4 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 15.4, the applicant provided the following:

Tier 2 Departure Not Requiring Prior NRC Approval

Administrative Departure

- STD DEP Admin

The applicant identifies two administrative departures: Subsection 15.4.2.1, "Features of the ABWR Automatic Thermal Limit Monitoring System (ATLM)," and Subsection 15.4.5.2.1.3, "Identification of Operator Actions."

COL License Information Items

- COL License Information Item 15.5 Mislocated Fuel Bundle Accident

This COL license information item addresses the mislocated fuel bundle accident. (COM 15.4-1)

- COL License Information Item 15.6 Misoriented Fuel Bundle Accident

This COL license information item addresses the misoriented fuel bundle accident. (COM 15.4-2)

15.4.3 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 reactivity and power distribution anomalies, and the associated acceptance criteria, are in Section 15.4.1 through Section 15.4.9.A of NUREG-0800.

In addition, 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 a Tier 2 departure. This departure does not require prior NRC approval and is subject to the requirements of 10 CFR Part 52, Appendix A, Section V.III.B.5, which are similar to the requirements in 10 CFR 50.59.

15.4.4 Technical Evaluation

As documented in NUREG–1503, the NRC staff reviewed and approved Section 15.4 of the certified ABWR DCD. The NRC staff reviewed Section 15.4 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.[1] The NRC 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 NRC staff reviewed the following information in the COL FSAR:

Tier 2 Departure Not Requiring Prior NRC Approval

Administrative Departure

STD DEP Admin

The applicant identified two administrative departures in FSAR Subsections 15.4.2.1 and 15.4.5.2.1.3. The first departure deletes Reference 15.4.1, which was also deleted in the DCD and is therefore acceptable.

The second administrative departure replaces Item 3, “Start up,” with “Shutdown.” The applicable position of the reactor mode switch is “shutdown,” not “startup.” This change corrects the DCD and is therefore acceptable.

The applicant defines administrative departures as minor corrections, such as editorial or administrative errors in the referenced ABWR DCD (e.g., misspellings, incorrect references, table headings, etc.). Administrative departures do not affect the presentation of any design discussion or the qualification of any design margin.

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 NRC staff determined it to be reasonable that this departure does not require prior NRC approval.

[1] See “*Finality of Referenced NRC Approvals*” in SER Section 1.1.3, for a discussion on the NRC 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 Items

- COL License Information Item 15.5 Mislocated Fuel Bundle Accident

This COL license information item addresses the mislocated fuel bundle accident. In Subsection 15.4.11.1 of the FSAR, Revision 3, the applicant stated, “The analysis results of the fuel bundle mislocated event will be prepared based on NRC approved methods and provided as an amendment to the FSAR in accordance with 10 CFR 50.71 (e) at least one year prior to fuel load. This analysis will reflect the final fuel design for the initial core loading. (COM 15.4-1).”

In RAI 04.04-2, the NRC staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. In its response to RAI 04.04-2, dated November 19, 2009 (ML093270045), the applicant stated that no departures are taken from the fuel design licensing basis that is described in the ABWR DCD, including the core loading map used for the transient and accident response analysis in DCD Figure 4.3-1 and the control rod strategy in DCD Table 4A-1. Because certified DCD Subsection 15.4.7.4, includes the mislocated fuel bundle accident analysis required for the core being licensed, COL License Information Item 15.5 is considered closed. Therefore, the applicant has withdrawn this commitment.

- COL License Information Item 15.6 Misoriented Fuel Bundle Accident

This COL license information item addresses the misoriented fuel bundle accident. In Subsection 15.4.11.2 of the FSAR, Revision 3, the applicant stated, “The analysis results of the fuel bundle misoriented event will be prepared based on NRC approved methods and provided as an amendment to the FSAR in accordance with 10 CFR 50.71 (e) at least one year prior to fuel load. This analysis will reflect the final fuel design for the initial core loading. (COM 15.4-2).”

In RAI 04.04-2, the NRC staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. In its response to RAI 04.04-2, dated November 19, 2009, the applicant stated that no departures are taken from the fuel design licensing basis that is described in the ABWR DCD, including the core loading map used for the transient and accident response analysis in DCD Figure 4.3-1 and the control rod strategy in DCD Table 4A-1. Since certified DCD Subsection 15.4.8.3 includes the misoriented fuel bundle accident analysis required for the core being licensed, COL License Information Item 15.6 is considered closed. Therefore, the applicant has withdrawn this commitment.

15.4.5 Post Combined License Activities

There are no post COL activities related to this section.

15.4.6 Conclusion

The NRC staff’s finding related to information incorporated by reference is in NUREG–1503. The NRC staff reviewed the application and checked the referenced DCD. The NRC 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 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the reactivity and power distribution anomalies that were incorporated by reference, have been resolved.

In addition, the NRC staff compared the additional information in the COLA to the relevant NRC regulations, the guidance in Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition," through Section 15.4.9.A, "Radiological Consequences of Control Rod Drop Accident (BWR)," of NUREG–0800. The NRC staff's review concluded that the applicant has adequately addressed COL License Information Items 15.5 and 15.6 in accordance with Section 15.4.1 through Section 15.4.9.A of NUREG–0800, and determined it to be reasonable that the identified Tier 2 departure is characterized as not requiring prior NRC approval per 10 CFR 52 Appendix A, Section VIII.B.5.

15.5 Increase in Reactor Coolant Inventory

This section of the FSAR addresses the AOOs that cause an increase in the RCS inventory. Increased inventory can be caused by an inadvertent high-pressure core flooder startup.

Section 15.5, "Increase in Reactor Coolant Inventory," of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Section 15.5, "Increase in Reactor Coolant Inventory," of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this subsection. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the AOOs that cause an increase in the reactor coolant inventory have been resolved.

15.6 Decrease in Reactor Coolant Inventory

15.6.1 Introduction

This section of the FSAR addresses the postulated accidents that cause a decrease in the reactor coolant inventory.

In order to consolidate the NRC staff's evaluation of the DBA radiological consequence assessments, this SER section evaluates Sections 15.6, "Decrease in Reactor Coolant Inventory," and 15.7, "Radioactive Release from Subsystems and Components," of the STP, Units 3 and 4, COL FSAR.

Subsection 15.6.5S, "Site-Specific Design Basis Accident Doses" of the FSAR describes the radiological consequence assessments of the DBAs for STP, Units 3 and 4, that use the site-specific atmospheric dispersion factors (χ/Qs) in STP FSAR Revision 12, Subsection 2.3S.4.2.1.1, "Offsite Dispersion Estimates," Tables 2.3S-23, "PAVAN Results - 0.5% χ/Q Values at the Dose Calculation EAB Site Exclusion Area Boundary Calculations - Building Wake Credit Is Not Included. Relative Concentration (χ/Q) Values (Sec/Cubic Meter) Versus Averaging Time (Note: Site Limit = 5% χ/Q Values)," 2.3S-24, "PAVAN Results - 0.5% χ/Q Values at the Dose Calculation LPZ Low Population Zone Calculations - Building Wake Credit Is Not Included. Relative Concentration (χ/Q) Values (Sec/Cubic Meter) Versus Averaging Time (Note: Site Limit = 5% χ/Q Values)," and 2.3S-25, "*ARCON96 χ /Q Values*

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

(sec/m^3)." The assessments are based on information in: (1) ABWR DCD Revision 4, Sections 15.6 and 15.7; and (2) STP FSAR Revision 12, Sections 15.6 and 15.7.

In ABWR DCD Sections 15.6 and 15.7, General Electric (GE) performed radiological consequence assessments of the following eight DBAs using the χ/Q values in ABWR DCD Tier 1, Table 5.0, "Site Parameters," and in ABWR DCD Tier 2, Table 15.6-13, "Loss of Coolant Accident Meteorology and Offsite Dose Results:"

- Failure of a small line carrying primary coolant outside the containment (DCD Section 15.6.2, "Failure of Small Line Carrying Primary Coolant Outside Containment").
- Steam system piping break outside the containment (DCD Section 15.6.4, "Steam System Piping Break Outside Containment").
- Loss-of-coolant accident (LOCA) inside the containment (DCD Section 15.6.5, "Loss-of-Coolant Accident (Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary)-Inside Containment").
- Cleanup water line break outside the containment (DCD Section 15.6.6, "Cleanup Water Line Break-Outside Containment").
- Radiological consequences of a radioactive gas waste system leak or failure (DCD Section 15.7.1, "Radiological Consequences of a Radioactive Gas Waste System Leak or Failure").
- Postulated radioactive release due to liquid radwaste tank failure (DCD Section 15.7.3, "Postulated Radioactive Release Due to Liquid Radwaste Tank Failure").
- Fuel handling accident (DCD Section 15.7.4, "Fuel-Handling Accident").
- Spent fuel cask drop accident (DCD Section 15.7.5, "Spent Fuel Cask Drop Accident").

15.6.2 Summary of Application

Sections 15.6 and 15.7 of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Sections 15.6 and 15.7 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Sections 15.6 and 15.7, the applicant provided the following:

Tier 1 Departures

- STD DEP T1 2.15-1 — Re-classification of Radwaste Building Substructure from Seismic Category I to Non-Seismic

This departure revises the seismic category of the radwaste building substructure from Seismic Category I to nonseismic. The applicant provided information in COL FSAR Subsection 15.7.3.1, "Identification of Cause and Frequency Classification," which identifies the radwaste building capabilities in preventing liquid releases.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 11.3-1 Gaseous Waste Management System

This departure makes a series of changes in the gaseous waste management system. The applicant provided supplemental information describing the potential of the gaseous waste management system to cause releases into the environment in COL FSAR Subsection 15.7.1.1, "Basis and Assumptions."

- STD DEP 15.6-1 Clean Up Water Line Break Meteorology and Dose Results

This departure makes changes to ABWR DCD Table 15.6-18, "Clean Up Water Line Break Meteorology and Dose Results." This departure was the result of an administrative error involving an incomplete markup of Revision 4 of the COL FSAR, which should have been made as part of the response to Environmental Report RAI 7.1-2, dated July 15, 2008 (ML082040684). The applicant also corrected an error in Revision 4 of the COL FSAR in the value for the third thyroid dose in Table 15.6-18. The departure is a standard departure to correct errors found in DCD Table 15.6-18, and is not dependent on site-specific input.

COL License Information Items

- COL License Information Item 15.7 Iodine Removal Credit

The applicant provided information addressing COL License Information Item 15.7 in FSAR Subsection 15.6.7.1. This COL license information item concerns the design characteristics of the main steamlines, drain lines, and main condensers for removing radioactive iodine before its release into the environment.

- COL License Information Item 15.9 Radiological Consequences of Non-line Break Accidents

The applicant provided information addressing COL License Information Item 15.9 in FSAR Subsection 15.7.6.1. This COL license information item concerns the radiological consequences of a postulated radioactive release due to a: (1) liquid radwaste tank failure accident, (2) fuel handling accident, and (3) fuel cask drop accident.

In the STP COLA, Part 5, "Emergency Plan," Section G.3, "Technical Support Center," provided information on the radiological consequence in the STP Technical Support Center (TSC) under reactor accident conditions. The TSC is the onsite technical support facility for emergency response. ABWR DCD Table 13.3-1, "ABWR Design Considerations for Emergency Planning Requirements," provided the ABWR design considerations for the TSC emergency planning requirements.

15.6.3 Regulatory Basis

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

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.

In addition, the relevant requirements of the Commission regulations and the associated acceptance criteria for reviewing the COL license information items are in Chapter 15 of NUREG-0800.

The ABWR design was certified as consistent with the dose reference values set forth in 10 CFR 100.11(a) in Subpart A, "Evaluation Factors for Stationary Power Reactor Site Applications before January 10, 1997 and for Testing Reactors," of 10 CFR Part 100, "Reactor Site Criteria." In STP, Units 3 and 4, COL FSAR, Chapter 15 the applicant incorporates by reference, the dose values set forth in 10 CFR 100.11(a) in lieu of the dose values set forth in 10 CFR 52.79 (a)(1).

RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Regulatory Position C.1.15.6.5, "Radiological Consequences," stated that the COL applicant should "show that site-specific short-term χ/Q values for the EAB, LPZ, and control room in Section 2.3S.4, "Short-Term Atmospheric Diffusion Estimates for Accident Releases," of the FSAR are within the χ/Qs assumed in the DCD."

The TSC is the onsite technical support facility for an emergency response required and specified by NRC regulation in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV.E.8. The TSC functional criteria are specified in NUREG–0696, "Functional Criteria for Emergency Response Facilities." Section 2.6 "Habitability," of NUREG-0696 requires the same radiological habitability as the main control room (MCR) under accident conditions meeting the dose acceptance criterion specified in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 19 "Control Room," and NUREG–0737, Supplement No. 1, "Clarification of Three Mile Island (TMI) Action Plan Requirements." GDC 19 and NUREG–0737 require radiological protection to assure that radiation exposure to any person working in the TSC does not exceed 0.05 Sievert (Sv) (5 roentgen equivalent man (rem)) whole body, or its equivalent to any part of the body, for the duration of an accident.

15.6.4 Technical Evaluation

As documented in NUREG–1503, the NRC staff reviewed and approved Sections 15.6 and 15.7 of the certified ABWR DCD. The NRC staff reviewed Sections 15.6 and 15.7 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 NRC staff's review confirmed that the information in the application and the information incorporated by reference, address the required information relating to these sections.

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

Compliance with siting criteria of 10 CFR 100.11(a) requires the applicant to show that for a plant located at the STP site, the radiological consequences of postulated accidents meet the following offsite radiological consequence evaluation factors:

1) *An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.*

2) *A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.*

Compliance with the control room habitability dose requirements of GDC 19 requires the applicant to show that for a plant located at the STP site, the control room provides adequate radiation protection to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) to the whole body or the equivalent dose to any part of the body for the duration of the accident.

In NUREG–1503, the NRC staff concluded that the site parameter χ/Qs proposed by GE for the EAB, the Low Population Zone (LPZ), and the control room of the ABWR plant — in conjunction with the engineered safety feature systems in the ABWR design — are sufficient to provide reasonable assurance that the radiological consequences of a postulated DBA will be within the dose reference values in 10 CFR Part 100.11(a) and GDC 19, respectively.

The DBA radiological consequence analyses in the ABWR DCD used site parameter χ/Q values. The site parameter χ/Q values are the only input to the DBA radiological consequence analyses for which corresponding site characteristics would result in site-specific doses that are different than the doses reported in the DCD. The applicant provided and discussed the STP site characteristic offsite χ/Q values in Subsection 2.3S.4.2.1.1, "Offsite Dispersion Estimates." In Section 2.3S.4, "Short-Term Diffusion Estimates," of this SER, the NRC staff discusses the review and evaluation of the STP site characteristic χ/Q values in Section 2.3S.4 of the STP, Units 3 and 4 ,COL FSAR, Revision 5.

In STP COL FSAR Revision 7, Table 15.6.5S-1, "Site-Specific χ/Q," the applicant compares the STP site characteristic offsite (EAB and LPZ) and control room χ/Q values with those values in ABWR DCD Tier 1, Table 5.0, "Site Parameters," and in ABWR DCD Tier 2, Tables 15.6-13 and 15.6-14, "Loss of Coolant Accident Meteorology and Control Room Dose Results."

The NRC staff's review determined that the STP site characteristic offsite χ/Q values are less than the comparable ABWR DCD χ/Q values. Therefore, the STP site-specific total dose for each DBA is less than the ABWR DCD generic total dose for each DBA. Also, the dose reference values specified in 10 CFR 100.11(a) are satisfied because the radiological consequence analysis for a DBA during any time period of radioactive material release from the plant is directly proportional to the χ/Q values for that time period.

For the STP site characteristic onsite control room χ/Q values, the NRC staff's review determined that the values exceeded the ABWR DCD generic control room χ/Q values for a turbine building release for two time intervals: 0 to 8 hours and 4 to 30 days. Therefore, the

NRC staff issued RAI 15.00.03-1 related to FSAR Section 15.6.5S, "Site-Specific Design Basis Accident Doses," requesting that the applicant provide the control room radiological consequence analyses for the DBAs that are relevant to the turbine building releases and to demonstrate that the STP, Units 3 and 4, control room still meets the dose acceptance criterion specified in GDC 19. The NRC staff tracked RAI 15.00.03-1 as Open Item 15.6-1 in the SER with open items.

In its response to RAI 15.00.03-1, dated November 30, 2009, (ML093360204), the applicant provided updated site-specific χ/Q values and control room doses. In Section 2.3S.4 of this SER, the NRC staff discusses the evaluation and acceptance of the STP site-specific χ/Q values provided in the response. The applicant's response to the RAI also stated that there are still two instances of the control room dose calculation where the site-specific χ/Q values exceed the value in the DCD. The two instances are the 4 to 30 day turbine building release and the 4 to 30 day reactor building release. Therefore, the applicant performed and provided the control room radiological consequence analyses for the turbine and reactor building releases following a DBA using the STP site-specific control room χ/Q values. The applicant's analyses showed that the control room doses for these two instances are still well within the dose acceptance criteria of 0.3 Sv (30 rem) thyroid, 0.05 Sv (5 rem) whole body, and 0.75 Sv (75 rem) beta skin dose as specified in SRP Section 6.4, "Control Room Habitability System," and in GDC 19. The NRC staff confirmed that the applicant's revised analysis was included in Revision 4 of the COL FSAR, and therefore the staff considers RAI 15.00.03-1 to be resolved and closed. Therefore, the NRC staff determined that the response to RAI 15.00.03-1 is acceptable and Open Item 15.6-1 is closed.

The NRC staff reviewed the following information in COL FSAR Sections 15.6 and 15.7:

Tier 1 Departures

The following Tier 1 departure identified by the applicant in FSAR Section 15.7 requires prior NRC approval, and the full scope of its technical impact may be evaluated in other sections of this SER accordingly. For more information, refer to the COLA, Part 7, Section 5.0 for a listing of all FSAR sections affected by this Tier 1 departure.

- STD DEP T1 2.15-1 — Re-classification of Radwaste Building Substructure from Seismic Category I to Non-Seismic

The NRC staff's evaluation of this departure is in Section 3.8, "Seismic Category I Structures," of this SER.

Tier 2 Departures Not Requiring Prior NRC Approval

The following Tier 2 departures not requiring prior NRC approval identified by the applicant in this section may also be evaluated in other sections of this SER accordingly. For more information, refer to the COLA, Part 7, Section 5.0 for a listing of all FSAR sections affected by these departures.

- STD DEP 11.3-1 — Gaseous Waste Management System

The NRC staff's evaluation of this departure is in Chapter Section 11.3, "Gaseous Waste Management System," of this SER.

- STD DEP 15.6-1 Clean Up Water Line Break Meteorology and Dose Results

The NRC staff evaluated the applicant's revisions to COL Revision 7, FSAR, Table 15.6-18 by performing independent calculations to confirm the applicant's revised dose results, and determined that the dose limits of 10 CFR 100.11(a) continue to be met. The revised dose results in FSAR Table 15.6-18 are made to correct an error in the reporting of the dose results in the DCD and are not a result of changes to the dose analysis inputs or assumptions from those evaluated by the NRC staff for the reference DCD as discussed in NUREG-1503. The revised dose results do not result in more than a minimal increase in the consequences of an accident previously evaluated in the DCD. 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 NRC staff determined it to be reasonable that this departure does not require prior NRC approval.

<u>COL License Information Items</u>

- COL License Information Item 15.7 Iodine Removal Credit

In Subsection 15.6.7.1 of STP FSAR Revision 3, the applicant addresses COL License Information Item 15.7, stating that:

> The design characteristics of the main steamlines, drain lines, and the main condenser are the same as specified in the reference ABWR DCD. As a result, the parameters in Table 15.6-8 Item II.D (MSIV leakage) and II.E (condenser data) remain unchanged. Since the iodine credit is a function of these parameters, the STP Units 3 & 4 iodine removal credit does not deviate from the referenced ABWR DCD.

The applicant has taken no departures from the parameters in Table 15.6-8, "Loss of Coolant Accident Parameters," Items II.D (MSIV leakage) and II.E (condenser data) and the iodine removal credit provided in the ABWR DCD. The NRC staff determined that the applicant's response to COL License Information Item 15.7, "Iodine Removal Credit," to be acceptable because the NRC staff reviewed and accepted the parameters in Table 15.6-8 and the iodine removal credit in NUREG-1503.

- COL License Information Item 15.9 Radiological Consequences of Non-Line Break Accidents

Table 15.7-7, "Radwaste System Failure Accident Meteorology and Dose Results;" Table 15.7-11, "Fuel-Handling Accident Meteorology Parameters And Radiological Effects;" and Table 15.7-13, "Fuel Cask Drop Accident Radiological Results – Meteorology and Dose Results;" of the certified ABWR DCD, provide radwaste system failure dose results, fuel handling accident dose results, and cask drop accident dose results, respectively, as a function of referenced χ/Q values in the ABWR DCD. The NRC staff reviewed and accepted these dose results in NUREG–1503.

The applicant used the site-specific χ/Q values to determine the resulting doses at the STP site for non-line break accidents in Subsection 15.7.6.1, using the relative ratio of the site characteristic χ/Q values to the site parameter χ/Q values. Subsequently, in its supplemental response to RAI 02.03.04-5, dated September 22, 2009, (ML 092230155), the applicant updated Subsection 15.7.6.1. These updates are based on the revised STP site-specific EAB short-term release χ/Q values which are evaluated and accepted by the NRC staff in Section 2.3S.4 of this SER. The updated values are included in Revision 5 of the STP, Units 3 and 4, COL FSAR.

The NRC staff determined that the information provided in response to COL License Information Item 15.9 is acceptable because the resulting doses for the radwaste system failure, fuel handling accident, and fuel cask drop accident are less than the referenced dose limit values specified in 10 CFR 100.11(a).

Technical Support Center Radiological Habitability

The NRC staff evaluated the dose analysis performed by the applicant which demonstrates that the TSC remains habitable, following postulated DBAs. As noted in SER Section 13.3, "Emergency Planning," the NRC staff issued RAI 13.03-73 requesting, among other items, that the applicant provide the TSC radiological consequence analyses for the DBAs to demonstrate that doses in the TSC meet the dose acceptance criterion of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) specified in SRP Section 15.0.3, "Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors." The NRC staff tracked RAI 13.03-73 as Open Item 15.6-2 in the SER with open items.

In its responses to RAI 13.03-73, dated June 1, 2010, (ML101550064) and June 21, 2010 (ML101750071), the applicant describes the dose to persons in the TSC during postulated DBAs. The applicant's results show that doses are well within the dose acceptance criterion of 0.05 Sv (5 rem) TEDE for the duration of an accident as specified in SRP Section 15.0.3. The NRC staff audited these calculations in June 2010 (ML102010690).

The NRC staff also issued RAIs 09.04.03-2 and 09.04.03-3, asking the applicant to explain why a design change for the emergency filter train, which involved upgrading charcoal filters from "at least 95%" removal efficiency (as noted in the ABWR DCD Tier 1, inspections, tests, analyses and acceptance criteria (ITAAC)) to filters with 99 percent removal efficiency, should not require an exemption request from the acceptance criteria in ABWR DCD Tier 1, Table 2.15.5m, "Service Building HVAC System," and site-specific ITAAC. In its response dated March 31, 2011 (ML110950045), the applicant explains that this design upgrade does not require a site specific ITAAC because these changes do not involve a change to or departure from Tier 1 information, Tier 2* information, or the TS; or require a license amendment under Paragraphs B.5.b or B.5.c of Section V.III in 10 CFR Part 52, Appendix A. Therefore, the use of 99 percent efficient filters is an improvement that does not require prior NRC approval. As described more fully below, the NRC staff audited the supporting calculations and noted several questions during its review (ML111590287). The applicant provided supplemental information dated June 16, 2011 (ML11173A182), which addressed the NRC staff's questions.

In particular, the NRC staff noted the following regarding the calculations: (1) the use of several radiological design criteria for the TSC; (2) the fact that the dose criterion of 0.25 Sv (25 rem) thyroid in Section G.3, "Technical Support Center," of the STP, Units 3 and 4, Emergency Plan would not be met for all DBAs; (3) an apparent discrepancy in particular flow rates described in the revised response to RAI 13.03-73 dated June 1, 2010 (ML101550064), as compared to the value used in the calculations; (4) a question of whether an air flow rate assumed for the service building (SB) clean area exhaust was appropriately conservative; and (5) a question of why site-specific ITAAC were not needed to meet the organ dose rate criterion specified in Section G.3

of the STP, Units 3 and 4, Emergency Plan. The applicant addressed these questions by providing supplemental information as follows:

1. The TSC design was approved by the NRC as part of the NRC's review of the ABWR standard design, and should have finality. Since the proposed changes to the SB Heating, Ventilation and Air Conditioning (HVAC) design increase the level of protection in the SB, Departure STD DEP 9.4-3 can be made without prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5.

2. The applicable dose-based regulatory requirements for design of the TSC are ITAAC specified in the DCD for shielding design (for direct radiation). These are 0.05 Sv (5 rem) to the whole body, or its equivalent, for the duration of the accident, and a design dose rate of less than 150 micro Sievert per hour (μSv/hr) (15 milli-rem [mrem] per hour) (averaged over 30 days) (ABWR DCD Tier 1, Table 3.2a, Item 3).

3. SRP Section 15.0.3, addresses TSC accident doses and stated an acceptance criterion of 0.05 Sv (5 rem) TEDE, which, unlike the ITAAC described above, addresses both direct radiation (deep dose equivalent) and intake (committed effective dose equivalent).

4. Flow rates described in the June 1, 2010, RAI response and the calculations are consistent. Though one item in the RAI response described the exhaust flow rate due only to SB clean area exhaust fans, another item also qualitatively explained that the normal mode exhaust rate from the SB clean area includes both building outleakage and mechanically-induced flow from the SB clean area exhaust fans. This response also addresses the question about whether the assumed emergency mode exhaust rate from the clean area is sufficiently conservative.

5. The proposed changes in the SB ventilation design are an improvement that does not require prior NRC approval in accordance with 10 CFR Part 52, Appendix A, Section VIII.B.5.

The NRC staff evaluated each of the responses and agreed with the applicant's submittals. The NRC staff also determined that all TSC radiological habitability dose calculations performed by Westinghouse for STP, Units 3 and 4, meet the dose acceptance criterion specified in SRP Section 15.0.3. Therefore, the NRC staff determined that the applicant's responses to RAI 13.03-73, RAI 09.04.03-2, and RAI 09.04.03-3 regarding the TSC radiological consequence analysis are acceptable. Therefore, the NRC staff determined that the RAIs are resolved and Open Item 15.6-2 is closed.

15.6.5 Post Combined License Activities

There are no post COL activities related to this section.

15.6.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. The NRC staff reviewed the application and checked the referenced DCD. The NRC 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 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the decrease in reactor coolant inventory that were incorporated by reference have been resolved.

In addition, the NRC staff compared the additional information in the COLA to the relevant NRC regulations, the guidance in Chapter 15 of NUREG–0800. The NRC staff's review concluded that the applicant has adequately addressed COL License Information Items 15.7 and 15.9 and the Tier 1 departure in accordance with Chapter 15 of NUREG–0800, and determined it to be 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.

15.7 Radioactive Release from Subsystems and Components

The NRC staff's evaluation of this section is in Chapter Section 15.6, "Decrease in Reactor Coolant Inventory," of this SER.

15.8 Anticipated Transients without Scram

This section of the FSAR addresses the potential failure of the reactor trip portion of the protection system to initiate a reactor scram when plant conditions call for a reactor scram to be initiated. The failure of the reactor to shut down during certain transients can lead to unacceptable RCS pressures, fuel conditions, and/or containment conditions. Therefore, alternate means to shut down the reactor must be addressed.

Section 15.8, "Anticipated Transients Without Scram," of STP COL FSAR incorporates by reference Section 15.8, "Anticipated Transients Without Scram," of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this subsection. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the anticipated transients without scram (ATWS) have been resolved.

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

15A Appendix 15A –Plant Nuclear Safety Operational Analysis (NSOA)

15A.1 Introduction

This FSAR appendix supports the single failure analysis required for Chapter 15.

15A.2 Summary of Application

Appendix 15A of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Appendix 15A of the certified ABWR DCD referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Appendix 15A, the applicant provided the following:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This departure pertains to the information in Figure 15A-7, "Safety System Auxiliaries-Group 2."

Tier 2 Departures Not Requiring Prior NRC Approval

Administrative Departures

- STD DEP Admin

This is an administrative departure pertaining to FSAR Figures 15A-13, -17, -19, -21, -25, -27, -29, -37, -38, -39, -40, -48, -51, -52, -53, -63, -64, -67, -68, -69, and -70.

- STD DEP Admin

This is an administrative departure pertaining to the rewording of text in FSAR Subsection 15A.6.2.3.11, "Control Rod Worth Control."

- STD DEP Admin

This is an administrative departure pertaining to the rewording of text in Subsection 15A.6.3.1, "General."

15A.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 plant NSOA, and the associated acceptance criteria, are in Chapter 15 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 V.III.A.4. Tier 2 departures not requiring prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section V.III.B.5, which are similar to the requirements in 10 CFR 50.59.

15A.4 Technical Evaluation

As documented in NUREG–1503, the NRC staff reviewed and approved Section 15A of the certified ABWR DCD. The NRC staff reviewed Appendix 15A 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 NRC staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this appendix.

The NRC staff reviewed the following information in the COL FSAR:

Tier 1 Departure

The following Tier 1 departure identified by the applicant in this section requires prior NRC approval, and the full scope of its technical impact may be evaluated in the other sections of this SER accordingly. For more information, refer to the COLA, Part 7, Section 5.0 for a listing of all FSAR sections affected by this Tier 1 departure.

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This departure removes the flammability control system and the associated ac and direct current (dc) power systems from Figure 15A-7. This departure is evaluated in Section 6.2.5, "Combustible Gas Control in Containment," of this SER. Since there is no flammability control system in STP, Units 3 and 4, in accordance with 10 CFR 50.44(c)(2), the NRC staff determined the deletion of the system in Figure 15A-7 to be acceptable.

Tier 2 Departures Not Requiring Prior NRC Approval

Administrative Departures

- STD DEP Admin

This departure changes the cross-reference of Figure 15A-13, "Protection Sequence for Loss of plant Instrument or Service Air System," from Figure 15A-67, "Commonality of Auxiliary Systems-Reactor Building Cooling Water System (RCWS)," to Figure 15A-64, "Protection Sequence for Core and Containment Cooling for Loss of Feedwater and Vessel Isolations." Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

This departure changes the cross-reference of Figure 15A-17,"Protection Sequences for Isolation of All Main Steamlines," from Figure 15A.6-57 to Figure 15A-64. Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

This departure changes the cross-reference of Figure 15A-19, "Protection Sequence for Loss of All Feedwater Flow," from Figure 15A-57, "Protection Sequence for Misplaced Fuel Bundle

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

Accident," to Figure 15A-64 and adds the word "flow" to the Event title. Within the review scope of this section, the NRC staff determined it to be reasonable that these editorial corrections do not require prior NRC approval.

This departure changes the cross-reference of Figure 15A-21, "Protection Sequence for Feedwater Controller Failure-Runout of One Feedwater Pump," from Figure 15A-19 to Figure 15A-24, "Protection Sequences for Main Turbine Trip, Bypass On," and changes "Nuclear Boiler System" to "Main Steam System." Within the review scope of this section, the NRC staff determined it to be reasonable that these editorial corrections do not require prior NRC approval.

This departure changes the cross-reference of Figures 15A-25, "Protection Sequences for Loss of Main Condenser Vacuum," and 15A-27, "Protection Sequences for Loss of Normal AC Power-Auxiliary transformer Failure," from Figure 15A-57 to Figure 15A-64. Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

In Figure 15A-29, "Protection Sequences for Inadvertent Opening of a Safety Relief Valve," this departure adds "automatic" in addition to a manual scram during suppression pool high temperature. Since there is an automatic scram during suppression pool high temperature, this addition is correct and acceptable. This departure also deletes "instrumentation" from the nuclear boiler system block. This is an editorial correction and hence is acceptable. This departure also changes the setpoint for automatic suppression pool cooling from 38 degrees Celsius (°C) to 35 °C (100 degrees Fahrenheit [°F] to 95 °F). The NRC staff issued RAI 15.08-3 requesting the applicant to justify the characterization of this change as administrative. The NRC staff tracked RAI 15.08-3 as Open Item 15A-1 in the SER with open item.

In its response to RAI 15.08-3, dated February 22, 2010 (ML100560113), the applicant submitted the revised version of Figure 15A-29 showing that the automatic start of suppression pool cooling remains at 38 °C (100 °F). The NRC staff reviewed the changes and determined them to be acceptable. RAI 15.08-3 is therefore resolved and Open Item 15.A-1 is closed.

The cross-reference of Figure 15A-37, "Protection Sequences for Loss of Coolant Piping Breaks in RCPB-Inside Containment," is changed from Figure 15A.6-32b to Figure 15A-38, "Protection Sequences for Loss of Coolant Piping Breaks in RCPB-Inside Primary Containment," and the word "Radiator" is changed to "Radiation." Within the review scope of this section, the NRC staff determined it to be reasonable that these editorial corrections do not require prior NRC approval.

The cross-reference of Figure 15A-38 is changed from Figure 15A.6-32a to Figure 15A-37. Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

The cross-reference of Figure 15A-39, "Protection Sequences for Liquid and Steam, Large and Small Piping Breaks Outside Containment," is changed from Figure 15A.6-33b to Figure 15A-40, "Protection Sequences for Liquid and Steam, Large and Small Piping Breaks Outside Primary Containment." Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

The cross-reference of Figure 15A-40 is changed from Figure 15A.6-33a to Figure 15A-39. Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

Figures 15A-48, 51, 52, 53, 64, 68, and 69, are within the review scope of this section, therefore, the NRC staff determined it to be reasonable that these editorial corrections do not require prior NRC approval.

In Figure 15A-63, “fine motion control rod drive (FMCRD)” is changed to “standby liquid control system (SLCS).” This change is acceptable since the applicable event is a reactor shutdown without control rods. Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

In Figure 15A-67, “high-pressure core flooder (HPCF) connected to Leak Detection and Isolation System” is deleted. There is no leak detection system for the HPCF. Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

The Figure 15A-70, “Commonality of Auxiliary Systems-Suppression Pool Storage,” cross-reference is changed from Tables 15A.6-2 through 15A.6-5 to Tables 15A-2, “Unacceptable Consequences Criteria Plant Event Category: Moderate Frequency Incidents (Anticipated operational Transients),” through 15A-5, “Capability Consequences Plant Event Category: Special Events.” Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

In Subsection 15A.6.2.3.11, “Control Rod Worth Control,” the basis for the limits imposed on the control rod pattern for low power condition is changed from “the control rod drop accident analysis” to “rod withdrawal error.” Since there is no control rod drop accident analysis in the DCD, replacing the control rod drop accident analysis with the applicable rod withdrawal error event. Within the review scope of this section, the NRC staff determined it to be reasonable that this editorial correction does not require prior NRC approval.

In Subsection 15A.6.3.1, “General,” the events that are applicable to moderate frequency incidents are revised to include events 23, 26, 27, 38 through 40, 44, 45, 48, and 49. Within the review scope of this section, the NRC staff determined it to be reasonable that these editorial corrections do not require prior NRC approval.

In general, the applicant defines administrative departures as minor corrections, such as editorial or administrative errors in the referenced ABWR DCD (e.g., misspellings, incorrect references, table headings, etc.). Administrative departures do not affect the presentation of any design discussion or the qualification of any design margin.

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 NRC staff determined it to be reasonable that this departure does not require prior NRC approval.

15A.5 Post Combined License Activities

There are no post COL activities related to this section.

15A.6 Conclusion

The NRC staff's finding related to information incorporated by reference, is in NUREG–1503. The NRC staff reviewed the application and checked the referenced DCD. The NRC 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 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the plant NSOA that were incorporated by reference have been resolved.

In addition, the NRC staff compared the additional information in the COLA to the relevant NRC regulations, the guidance in Chapter 15 of NUREG–0800. The NRC staff's review concluded that the applicant has adequately addressed the identified Tier 1 in accordance with the Chapter 15 of NUREG–0800, and determined it to be reasonable that the identified Tier 2 departure is characterized as not requiring prior NRC approval per 10 CFR 52 Appendix A, Section VIII.B.5.

15B Appendix 15B–Failure Modes and Effects Analysis (FMEA)

15B.1 Introduction

This FSAR appendix supports the single-failure analysis required for Chapter 15.

15B.2 Summary of Application

Appendix 15B of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Appendix 15B of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Appendix 15B, the applicant provided the following:

Tier 1 Departure

- STP DEP T1 3.4-1 Safety-Related (I&C) Architecture

This departure describes changes to the instrumentation and control (I&C) architecture and nomenclature. In this appendix, this departure affects Table 15B-3, "DCF of the RTIS and the ELCS," and Sections 15B.1 and 15B.4, by deleting the words "Essential Multiplexing System," in the title and renaming it as "Data Communication Function of the Reactor Trip and Isolation System (RTIS) and ESF Logic and Control System (ELCS)."

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 7.7-1 RPV Water Level Instrumentation

This departure clarifies reactor pressure vessel (RPV) water level instrument lines with a condensing chamber, and shows revisions to Figure 15B-1 and to Section 15B.2.3 that indicate the nuclear boiler instrument line fill system.

15B.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 single-failure analysis, and the associated acceptance criteria, are in Chapter 15 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, “Changes, tests, and experiments.”

15B.4 Technical Evaluation

As documented in NUREG–1503, the NRC staff reviewed and approved Appendix 15B of the certified ABWR DCD. The NRC staff reviewed Appendix 15B 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 NRC staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this appendix.

The NRC staff reviewed the following information in the COL FSAR:

Tier 1 Departure

The following Tier 1 departure identified by the applicant in this section requires prior NRC approval, and the full scope of its technical impact may be evaluated in the other sections of this SER accordingly. For more information, refer to the COLA Part 7, Section 5.0 for a listing of all FSAR sections affected by this Tier 1 departure.

- STD DEP T1 3.4-1 Safety-Related I&C Architecture

In Appendix 15B, STD DEP T1 3.4-1 refers to Table 15B-3, and makes an editorial change. This change is in the data communication function of the RTIS and the ELCS, and the evaluation is addressed in Chapter 7 of this SER. Failure modes, an effects analysis (FMEA), and the common-cause failure (CCF) of RTIS and ELCS data communication functions are evaluated in Chapter 19 of this SER.

Tier 2 Departure Not Requiring Prior NRC Approval

The following Tier 2 departure not requiring prior NRC approval identified by the applicant in this section may also be evaluated in other sections of this SER accordingly. For more information, refer to COLA Part 7, Section 5.0 for a listing of all FSAR sections affected by this departure.

- STD DEP 7.7-1 RPV Water Level Instrumentation

In Appendix 15B, STD DEP 7.7-1 refers to changes in Figure 15B-1, “Simplified Control Rod Drive (CRD) System Process Diagram.” This figure is revised to indicate the nuclear boiler instrument line fill system. The CRD system supplies water to the reactor vessel level-monitoring system and hence the cross-tie shown between the CRD system and the nuclear

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

boiler system. Within the review scope of this section, the NRC staff determined it to be reasonable that this departure does not require prior NRC approval. This departure is further addressed in Chapter 7 of this SER.

15B.5 Post Combined License Activities

There are no post COL activities related to this section.

15B.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. The NRC staff reviewed the application and checked the referenced DCD. The NRC 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 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the FMEA that were incorporated by reference have been resolved.

In addition, the NRC staff compared the information in the COLA to the relevant NRC regulations, the guidance in Chapter 15 of NUREG–0800. The NRC staff's review concluded that the applicant has adequately addressed the identified Tier 1 in accordance with the Chapter 15 of NUREG–0800, and determined it to be reasonable that the identified Tier 2 departure is characterized as not requiring prior NRC approval per 10 CFR 52 Appendix A, Section VIII.B.5.

15C Appendix 15C–Not used

15D Appendix 15D–Probability Analysis of Pressure Regulator Downscale Failure

This FSAR appendix supports the single-failure analysis required for Chapter 15.

Appendix 15D of the STP COL FSAR, Revision 12, incorporates by reference Appendix 15D, "Probability Analysis of Pressure Regulator Downscale Failure," of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this subsection. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the pressure regulator downscale failure have been resolved.

15E Appendix 15E–ATWS Performance Evaluation

15E.1 Introduction

This FSAR appendix is supplementary information supporting ATWS performance evaluation.

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

15E.2 Summary of Application

Appendix 15E of the STP, Units 3 and 4, COL FSAR, Revision 12, incorporates by reference Appendix 15E of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Appendix 15E, the applicant provided the following:

Tier 1 Departure

- STD DEP T1 3.4-1 Safety-Related I&C Architecture

This departure identifies changes to DCD Figures 15E-1a, 15E-1b, and 15E-1c.

Tier 2 Departures Not Requiring Prior NRC Approval

Administrative Departures

- STD DEP Admin

In FSAR Revision 2, Section 15E.4, "ATWS Logic and Setpoints," the applicant provided a revised discussion for the automated initiation of the automatic depressurization system (ADS) inhibit.

- STD DEP Admin

In Section 15.E.5, "Selection of Events," the applicant provided clarifications and editorial changes.

15E.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 ATWS performance evaluation, and the associated acceptance criteria, are in Chapter 15 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.

15E.4 Technical Evaluation

As documented in NUREG–1503, the NRC staff reviewed and approved Appendix 15E of the certified ABWR DCD. The NRC staff reviewed Appendix 15E 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 NRC staff's review

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

confirmed that the information in the application and the information incorporated by reference address the required information relating to this appendix.

The NRC staff reviewed the following information in the COL FSAR:

Tier 1 Departure

The following Tier 1 departure identified by the applicant in this section requires prior NRC approval, and the full scope of its technical impact may be evaluated in other sections of this SER accordingly. For more information, refer to the COLA, Part 7, Section 5.0 for a listing of all FSAR sections affected by this Tier 1 departure.

- STD DEP T1 3.4-1 Safety-Related I& C Architecture

The NRC staff's detailed evaluation of this departure is in Chapter 7 of this SER. However, the following RAI was issued in order to understand changes made in Appendix 15E as a result of this departure:

The applicant revised the logic in the STP, Units 3 and 4, FSAR Figure 15E-1a, "ATWS Mitigation Logic ([alternate rod insertion] ARI, FMCRD Run-In, RPT, Manual Initiation);" Figure 15E-1b, "ATWS Mitigation Logic (SLCS Initiation, Feedwater Runback);" and Figure15E-1c, "ATWS Mitigation Logic (SLCS Initiation, Feedwater Runback);" based on Departure STD DEP T1 3.4-1 and Tier 2 administrative departures. However, no changes were made to the ATWS logic-related text in the FSAR, specifically Sections 2.2.8 and 3.4, "Instrumentation and Control," in FSAR Tier 1 and Sections 7.4, "Systems Required for Safe Shutdown," and 15E.1 in FSAR Tier 2. No explanation of the logic changes shown on these revised FSAR figures was provided in either of the cited departures. The NRC staff also noted that some of the changes made to these FSAR figures are not annotated (back circled). Due to the above stated reasons, the NRC staff was unable to verify the acceptability of the departed ATWS mitigation logic. Therefore, the NRC staff issued RAI 15.08-2 requesting additional clarifications. The NRC staff tracked RAI 15.08-2 as Open Item 15E-1 in the SER with open items.

In its response to RAI 15.08-2, dated March 30, 2010 (ML100920022), the applicant updated FSAR Figure 15E-1b, "ATWS Mitigation Logic (SLCS Initiation, Feedwater Runback," so that the ATWS mitigation logic now remains unchanged from the one described in the certified ABWR DCD. Changes to FSAR Figures 15E-1a, "ATWS Mitigation Logic (ARI, FMCRD Run-In, RPT, Manual Initiation)," 15E-1b, and 15E-1c, "ATWS Mitigation Logic (SLCS Initiation, Feedwater Runback)," are now related to nomenclature changes identified in Departure STD DEP T1 3.4-1 and to some administrative editorial changes. The NRC staff reviewed this RAI response and determined that the proposed FSAR changes do not impact the ATWS mitigation logic described in the ABWR DCD. The safety system logic and control nomenclature-related changes made in accordance with Departure STD DEP T1 3.4-1 are evaluated in Chapter 7 of this SER. The RAI response is therefore acceptable, and Open Item 15.E-1 is closed.

Tier 2 Departures Not Requiring Prior NRC Approval

Administrative Departures

- STD DEP Admin

In FSAR Revision 2, Section 15E.4, “ATWS Logic and Setpoints,” the applicant provided a revised discussion for the automated initiation of the ADS inhibit. The NRC staff’s review identified the following inconsistency and issued RAI 15.08-1:

> In the certified DCD, automated initiation of ADS is inhibited unless there is a coincident low reactor water level signal (level 1.5) and an average power range monitor (APRM) ATWS permissive signal. In Revision 2 of the COL FSAR, an administrative change was made to indicate the deletion of the low reactor water level signal (level 1.5) from the ADS inhibit logic during ATWS. Justify the deletion of the reactor water level from the logic.

In its response to RAI 15.08-1, dated July 2, 2009 (ML091880283), the applicant confirms that there is no change to the ATWS ADS Inhibit mitigation function from that described in the DCD and the logic that is shown in Figure 15E-1c. This administrative change was deleted in Revision 3 of the COL FSAR. Therefore, this RAI is 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 NRC staff determined it to be reasonable that this departure does not require prior NRC approval.

- STD DEP Admin

The NRC staff reviewed the administrative changes in Section 15E.5 of the COL FSAR. The NRC staff determined:

1. Addition of “boron injection” in the “Loss of Feedwater” event is acceptable since the injection function is an ATWS mitigation function.

2. In the “Turbine Trip with Bypass Valves Open” event, the change from “neutron flow heat” to “neutron flux” is editorial and hence is acceptable. Also, the deletion of “vessel pressure” is acceptable since during a turbine trip with the bypass valves open, the reactor pressure will decrease due to the opening of the turbine bypass valves. This is a DCD correction and hence 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 NRC staff determined it to be reasonable that this departure does not require prior NRC approval.

15E.5 Post Combined License Activities

There are no post COL activities related to this section.

15E.6 Conclusion

The NRC staff's finding related to information incorporated by reference, is in NUREG–1503. The NRC staff reviewed the application and checked the referenced DCD. The NRC 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 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the ATWS performance evaluation that were incorporated by reference, have been resolved.

In addition, the NRC staff compared the additional information in the COLA to the relevant NRC regulations, the guidance in Section 15 of NUREG–0800. The NRC staff's review concluded that the applicant has adequately addressed the Tier 1 departure in accordance with Chapter 15 of NUREG–0800, and determined it to be 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.

15F Appendix 15F–LOCA Inventory Curves

The NRC staff's evaluation of this appendix is included in Chapter Section 6.3, "Emergency Core Cooling Systems," of this SER.