15.0 TRANSIENT AND ACCIDENT ANALYSES

15.0.1 Introduction

This section of 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 of the COL FSAR 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, Appendix A, with no departures or supplements. In addition, in FSAR Section 15.0, the applicant provides the following:

COL License Information Items

- COL License Information Item 15.1 Anticipated Operational Occurrences

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

- 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) (FSER related to the certified ABWR DCD).

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

The regulatory basis for reviewing COL license information items are in Section 15 of NUREG–0800.

15.0.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 15.0 of the certified ABWR DCD. The 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 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 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 states, "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 states, "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 states, "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 1 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 staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. The applicant submitted the response to RAI 04.04-2 in a letter dated November 19, 2009 (ML093270045). The applicant

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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.

states 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 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 commitments COM 15.0-1 through 15.0-3 are closed. Therefore, the applicant has withdrawn these commitments. Verification that the applicant's proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15.0-1**.

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. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information. With the exception of Confirmatory Item 15.0-1, there is no outstanding information 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 transient and accident analyses that were incorporated by reference have been resolved.

However, as a result of the Confirmatory Item 15.0-1, the staff was unable to finalize the conclusions relating to this section, in accordance with the NRC requirements.

15.1 Decrease in Reactor Coolant Temperature

15.1.1 Introduction

This section of the FSAR addresses the AOO 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 of the COL FSAR incorporates by reference Section 15.1 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures. In addition, in FSAR Section 15.1, the applicant provides 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, the applicant provides instrument response times for STP Units 3 and 4.

15.1.3 Regulatory Basis

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

In addition, the relevant requirements of the Commission's regulations, and the associated acceptance criteria, for the decrease in reactor coolant temperature are in Section 15.1.1-15.1.4 of NUREG–0800.

In addition, the relevant requirements of the Commission's regulations, and the associated acceptance criteria for reviewing supplemental information are in Section 15.1.1-15.1.4 of NUREG–0800.

15.1.4 Technical Evaluation

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

Supplemental Information

In response to a COL license information item in ABWR DCD Subsection 15.1.2.3.2.2, the applicant provides 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).

In RAI 04.04-2, the staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. The applicant submitted the response to RAI 04.04-2 in a letter dated November 19, 2009 (ML093270045). In this response, the applicant states that no

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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.

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, COM 15.1-1 is considered closed. Therefore, the applicant has withdrawn this commitment. Verification that the applicant's proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15.1-1**.

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 staff issued RAI 15.01.01-15.01.04-1. In the response to this RAI dated July 2, 2009 (ML091880283), the applicant states 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 found acceptable in NUREG–1503, the staff considers this RAI 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. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information. With the exception of Confirmatory Item 15.1-1, there is no outstanding information 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 decrease in reactor coolant temperature that were incorporated by reference have been resolved.

However, as a result of the Confirmatory Item 15.1-1, the staff was unable to finalize the conclusions relating to this section, in accordance with the NRC requirements.

15.2 Increase in Reactor Pressure

15.2.1 Introduction

This section of the FSAR addresses the AOO 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 of the STP Units 3 and 4 COL FSAR 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 provides 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 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 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’s regulations, and the associated acceptance criteria, for the increase in reactor pressure are in Section 15.2.1-15.2.5 of NUREG–0800.

In addition, the relevant requirements of the Commission’s regulations, and the associated acceptance criteria, for reviewing COL license information items and supplemental information are in NUREG–0800, Section 15.2.1-15.2.5.

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, NRC staff reviewed and approved Section 15.2 of the certified ABWR DCD. The 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 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 reviewed the 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 COL application, Part 07, Section 5.0 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 staff's review of the radiological aspects of this departure is in Section 11.5 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 SER accordingly. For more information, refer to the COL application, Part 07, 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 staff's review of this departure is in Chapter 8 of this SER.

[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.

Supplemental Information

In 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 provides 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 COM15.2-2)

In RAI 04.04-2, the staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. The applicant submitted the response to RAI 04.04-2 in a letter dated November 19, 2009 (ML093270045). In this response, the applicant states 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 staff found 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, and RAI 04.04-2 is considered closed. Therefore, the applicant has withdrawn these commitments. Verification that the applicant's proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15.2-1**.

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 staff determined that this event, according to SRP Section 15.0.3, is not a DBA. The radiological impact of this transient involves no fuel damage. 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, "Radiological Effects of MSIV Closures," of the STP FSAR, Revision 3, states that the radiological consequences of the inadvertent MSIV closure are 9.9 E-5 milli-Gray (mGy) for thyroid dose and 2.9 E-3 mGy whole body dose. Subsequently, in a supplemental response to RAI 02.03.04-5 dated September 22, 2009 (ML092230155), the applicant updated Subsection 15.2.10.1 to show 4.4 E-4 milli-Gray (mGy) for thyroid dose and 1.3 E-2 mGy whole body dose. These updates are based on the revised STP site-specific EAB long-term release χ/Q values which are evaluated and accepted by the staff in Section 2.3S.5 of this SER. Revision 3 of the STP Unit 3 and 4 COL FSAR does not capture the updated FSAR changes.

The staff determined that these dose values are based on: (1) the radiation doses provided in Table 15.2-12, "Dose Evaluation and Meteorology," 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 staff in its review of the ABWR standard reactor design certification. Therefore, the staff found that the radiological consequences of the inadvertent

MSIV closure provided by the applicant are consistent with the values that were reviewed and approved by the staff in NUREG–1503. Verification that the applicant's proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15.2-2**.

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. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information. With the exception of Confirmatory Items 15.2-1 and 15.2-2, there is no outstanding information 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 increase in reactor pressure that were incorporated by reference have been resolved.

However, as a result of the Confirmatory Items 15.2-1 and 15.2-2, the staff was unable to finalize the conclusions relating to this section, in accordance with the NRC requirements.

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

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

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

Section 15.3 of the STP Units 3 and 4 COL FSAR incorporates by reference, with no departures or supplements, Section 15.3, "Decrease in Reactor Coolant System Flow Rate," of the certified ABWR DCD, Revision 4, which is referenced in 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 subsection. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the AOO that cause a decrease in the reactor RCS flow rate have been resolved.

15.4 <u>Reactivity and Power Distribution Anomalies</u>

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:

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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.

- 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 of the STP Units 3 and 4 COL FSAR 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 provides the following:

Tier 2 Departure Not Requiring Prior NRC Approval

Administrative Departure

- STP 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’s regulations, and the associated acceptance criteria, for the reactivity and power distribution anomaliesare in Section 15.4 of NUREG–0800.

The relevant requirements for the Commission’s regulations and the associated acceptance criteria for reviewing COL license information items is in Section 15.4 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, NRC staff reviewed and approved Section 15.4 of the certified ABWR DCD. The 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 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 reviewed the information in the FSAR:

Tier 2 Departure Not Requiring Prior NRC Approval

Administrative Departure

STD DEP Admin

The applicant identified two administrative departures: Section 15.4.2.1, "Features of the ABWR Thermal Limit Monitoring System," and Section 15.4.5.2.1.3, "Identification of Operator Actions." 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.

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. The applicant states in Subsection 15.4.11.1 of the FSAR, "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 staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. The applicant submitted the response to RAI 04.04-2 in a letter dated November 19, 2009 (ML093270045). In this response, the applicant states 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. Verification that the applicant's proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15.4-1**.

[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.

- COL License Information Item 15.6 Misoriented Fuel Bundle Accident

This COL license information item addresses the misoriented fuel bundle accident. The applicant states in Subsection 15.4.11.2 of the FSAR, "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 staff informed the applicant that the method proposed by the applicant is not an acceptable resolution. The applicant submitted the response to RAI 04.04-2 in a letter dated November 19, 2009. in this response, the applicant states 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. Verification that the applicant's proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15.4-2**.

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. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information. With the exception of Confirmatory Items 15.4-1 and 15.4-2, there is no outstanding information 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 reactivity and power distribution anomalies that were incorporated by reference have been resolved.

However, as a result of the Confirmatory Items 15.4-1 and 15.4-2, the staff was unable to finalize the conclusions relating to this section, in accordance with the NRC requirements.

15.5 Increase in Reactor Coolant Inventory

15.5.1 Introduction

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

Section 15.5 of the STP Units 3 and 4 COL FSAR incorporates by reference, with no departures or supplements, Section 15.5, "Increase in Reactor Coolant Inventory," of the certified ABWR DCD, Revision 4, which is referenced in 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 subsection. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the AOO 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 describes 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 3, Section 2.3S.4.2.1.1, "Offsite Dispersion Estimates"; Tables 2.3S.23, 2.4S.24, and 2.3S-25.

The assessments are based on information in (1) ABWR DCD Revision 4 Chapter 15, Section 15.6, "Decrease in Reactor Coolant Inventory"; and (2) STP FSAR Revision 3 Chapter 15, Section 15.6, "Decrease in Reactor Coolant Inventory."

In ABWR DCD Section 15.6, "Decrease in Reactor Coolant Inventory," and in Section 15.7, "Radioactive Release from Subsystems and Components," General Electric (GE) performed radiological consequence assessments of the following six DBAs using the hypothetical set of χ/Q values in ABWR DCD Tier 1 Table 5.0, "Site Parameters," and in ABWR DCD Tier 2, Table 15.6-13:

- Failure of a small line carrying primary coolant outside the containment (DCD Section 15.6.2)
- Steam system piping break outside the containment (DCD Section 15.6.4)
- Loss of coolant accident inside the containment (DCD Section 15.6.5)
- Cleanup water line break outside the containment (DCD Section 15.6.6)
- Fuel handling accident (DCD Section 15.7.4)
- Spent fuel cask drop accident (DCD Section 15.7.5)

This SER section evaluates Sections 15.6 and 15.7 of the STP Units 3 and 4 COL FSAR.

15.6.2 Summary of Application

Sections 15.6 and 15.7 of the STP Units 3 and 4 COL FSAR incorporate 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 provides the following:

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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.

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 provides supplemental information in COL FSAR Subsection 15.7.3.1, “Identification of Causes and Frequency Classification,” which identifies the radwaste building capabilities in preventing liquid releases.

Tier 2 Departures Not Requiring Prior NRC Approval

- STP DEP 11.3-1 — Gaseous Waste Management System

This departure makes a series of changes in the gaseous waste management system. The applicant provides 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.”

COL License Information Items

- COL License Information Item 15.7 — Iodine Removal Credit

The applicant provides 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 steam lines, 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 provides 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 COL application, Part 5, “Emergency Plan,” Section G.3, “Technical Support Center,” provides 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 an emergency response. ABWR DCD Table 13.3-1 provides the ABWR design considerations for the TSC emergency planning requirements.

15.6.3 Regulatory Basis

The regulatory basis for reviewing 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 for the Commission's regulations and the associated acceptance criteria for reviewing COL license information items is in Section 15.0.3 of NUREG-0800 and in RG 1.183.

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, 1997and for Testing Reactors," of 10 CFR Part 100, "Reactor Site Criteria." In STP Units 3 and 4 COL FSAR Chapter 15, "Accident and Analysis," 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," Regulatory Position C.III.1, Subsection C.1.15.6.5, "Radiological Consequences," states 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 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 10 CFR Part 50 Appendix E, 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 under accident conditions meeting the dose acceptance criterion specified in 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 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 5 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, NRC staff reviewed and approved Sections 15.6 and 15.7 of the certified ABWR DCD. The 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 staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to these sections.

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:

(A) An individual located at any point on the boundary of the exclusion area for two hours immediately following the onset of the postulated fission product release, would not

[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.

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.

(B) *An individual located at any point on the outer boundary of the low-population zone, 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–503, the staff concluded that the bounding χ/Qs proposed by GE for the EAB, the 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 the hypothetical set of χ/Q values in place of site-specific values. The χ/Q values are the only input to the DBA radiological consequence analyses that are impacted by the site characteristics. The applicant provides and discusses the STP site-specific 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 staff discusses the review and evaluation of the STP site-specific χ/Q values in Section 2.3S.4 of the STP Units 3 and 4 COL FSAR, Revision 3.

The estimated offsite DBA dose calculated for a particular site is impacted by the site characteristics through the calculated χ/Q input to the radiological analysis. The resulting site-specific dose would be different from the resulting dose calculated generically for the ABWR design. All other inputs and assumptions in the radiological consequence analyses remain the same as in the DCD. Smaller χ/Q values are associated with a greater dilution capability that results in lower radiological doses. When comparing a DCD generic site χ/Q value with a site-specific χ/Q value, the site is acceptable for the design if the site-specific χ/Q value is smaller than the DCD generic site χ/Q value. Such a comparison shows that the site has better dispersion characteristics than the reactor design requires.

In the STP FSAR Revision 3, Table 15.6.5S-1, “Site-Specific χ/Q,” the applicant compares the STP site-specific 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.

The staff finds that the STP site-specific 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-specific onsite control room χ/Q values, the staff's review found 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 staff issued RAI 15.00.03-1 related to FSAR Section 15.6.5S, "Site-Specific Design Basis Accident Doses," requesting the applicant to 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 staff tracked RAI15.00.03-1 as Open Item 15.6-1 in the SER with open item.

In the response to RAI 15.00.03-1 dated November 30, 2009 (ML093360204), the applicant provides updated site-specific χ/Q values and control room doses. In Section 2.3S.4 of this SER, the staff discussed the evaluation and acceptance of the STP site-specific χ/Q values provided in the response. The applicant's response to the RAI also states 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 30 rem thyroid, 5 rem whole body, and 75 rem beta skin dose as specified in SRP Section 6.4 and in GDC 19. The staff performed independent confirmatory calculations and confirmed the applicant's results. Therefore, the staff determined that the response to RAI 15.00.03-1 is acceptable and Open Item 15.6-1 is closed. Verification that the applicant's proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15.6-1**.

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

Tier 1 Departures

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 COL application, Part 07, 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 staff's evaluation of this departure is in Sections 3.8 and 12.2 of this SER.

Tier 2 Departures 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 the COL application, Part 07, Section 5.0 for a listing of all FSAR sections affected by this departure.

- STP DEP 11.3-1 — Gaseous Waste Management System

The staff's evaluation of this departure is in Section 11.3 of this SER.

COL License Information Items

- 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,"Iodine Removal Credit," 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, Items II.D (MSIV leakage) and II.E (condenser data) and the iodine removal credit provided in the ABWR DCD. The staff found that the applicant's response to COL License Information Item 15.7, "Iodine Removal Credit," is acceptable because the staff reviewed and accepted the parameters in Table 15.6-8, "Loss-of-Coolant Accident Parameters," 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. 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-specific χ/Q values to the referenced χ/Q values. Subsequently, in a supplemental response to RAI 02.03.04-5 dated September 22, 2009 (ML092230155), 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 staff in Section 2.3S.4 of this SER. Revision 3 of the STP Unit 3 and 4 COL FSAR does not capture the updated site-specific χ/Q values or site-specific EAB doses. Verification of changes to FSAR Subsection 15.7.6.1 is being tracked as Confirmatory Item 15.6-1.

The staff performed independent dose calculations and confirmed the applicant's dose results. Therefore, the 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.

In the STP COL application, Part 5, "Emergency Plan," Section G.3, "Technical Support Center," the applicant states:

> Each Technical Support Center is provided sufficient radiological protection and monitoring equipment to assure that radiation exposure to any person working in the activated Technical Support Center will not exceed five (5) rem TEDE or twenty-five (25) rem thyroid CDE during the duration of a declared accident.

In SER Section 13.3, the staff issued RAI 13.03-73 requesting, among other items, that the applicant provide the TSC radiological consequence analyses for the DBAs to demonstrate it meets the dose acceptance criterion of 5 rem TEDE specified in SRP Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Water Reactors." The staff tracked RAI13.03-73 as Open Item 15.6-2 in the SER with open item.

In the revised response to RAI 13.03-73 dated June 1, 2010 (ML101550064), the applicant provides the results of radiological consequence analyses for the TSC habitability under postulated DBAs. The applicant's results show that they are well within the dose acceptance criterion of 5 rem TEDE for the duration of an accident as specified in SRP Section 15.0.3.

On June 25, 2010, the staff performed an audit at the Westinghouse Offices in Rockville, Maryland, to review the STP and Westinghouse dose calculations and related assumptions used in the STP Units 3 and 4 TSC radiological habitability analyses. The staff found that the calculations were based on the fission product releases listed in the certified ABWR DCD. The calculations were performed using an NRC computer code provided in NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," Version 3.0.3. RADTRAD computer code was developed by the Sandia National Laboratories for the NRC and it calculates the radionuclide transport and removal and dose at selected receptors.

The staff found that all TSC radiological habitability dose calculations performed by Westinghouse for STP Units 3 and 4 were in accordance with SRP Section 15.0.3 and the guidelines provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accident at Nuclear Power Reactors," meeting the dose acceptance criterion specified in SRP Section 15.0.3. Therefore, the staff determined that the applicant's revised response to RAI 13.03-73 regarding the TSC radiological consequence is acceptable and Open Item 15.6-2 is closed. Verification that the applicant's proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15.6-2**.

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. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information. With the exception of Confirmatory Items 15.6-1 and 15.6-2, 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 decrease in reactor coolant inventory that were incorporated by reference have been resolved.

However, as a result of the Confirmatory Items 15.6-1 and 15.6-2, the staff was unable to finalize the conclusions relating to this section, in accordance with the NRC requirements.

15.7 Radioactive Release from Subsystems and Components

The staff's evaluation of this section is in Section 15.6 of this SER.

15.8 Anticipated Transients without Scram

15.8.1 Introduction

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 of STP COL FSAR incorporates by reference, with no departures or supplements, Section 15.8, "Anticipated Transients Without Scram," of the certified ABWR DCD, Revision 4, which is referenced in 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 subsection. Pursuant to 10 CFR 52.63(a)(5) and 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 staff's review related to verification of the scope of information to be included in a COL application that references a design certification.

Evaluation of Chapter 15 Appendices

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

15A.1 Introduction

This 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 incorporates by reference Appendix 15A of the certified ABWR DCD referenced in 10 CFR Part 52, Appendix A. In addition, in Appendix 15A, the applicant provides the following departures:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This departure pertains to the information in Figure 15A-7.

Tier 2 Departures Not Requiring Prior NRC Approval

Administrative Departures

- STD DEP Admin

This is an administrative departure pertaining to 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 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 Subsection15A.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’s regulations for the plant NSOA, and the associated acceptance criteria, are in Section 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, NRC staff reviewed and approved Section 15A of the certified ABWR DCD. The 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 staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this appendix.

In addition, the staff reviewed the following:

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 COL application, Part 07, 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 DC power systems from Figure 15A-7. This departure is evaluated in Section 6.2.5 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 staff found the deletion of the system in Figure 15A-7 acceptable.

Tier 2 Departures Not Requiring Prior NRC Approval

Administrative Departures

- STD DEP Admin

This departure changes the cross-reference of Figure 15A-13- from Figure 15A-67 to Figure 15A-64. This is an editorial correction and hence is acceptable.

This departure changes the cross-reference of Figure 15A-17 from Figure 15A.6-57 to Figure 15A-64. This is an editorial correction and hence is acceptable.

This departure changes the cross-reference of Figure 15A-19 from Figure 15A-57 to Figure 15A-64 and adds the word “flow” to the Event title. These are editorial corrections and hence are acceptable.

This departure changes the cross-reference of Figure 15A-21 from Figure 15A-19 to Figure 1A-24 and changes “Nuclear Boiler System” to “Main Steam System.” These are editorial corrections and hence are acceptable.

[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.

This departure changes the cross-reference of Figures 15A-25 and 15A-27 from Figure 15A-57 to Figure 15A-64. This is an editorial correction and hence is acceptable.

In Figure 15A-29, 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 C to 35 degrees C. The NRC issued RAI 15.08-3 requesting the applicant to justify the characterization of this change as administrative. The staff tracked RAI15.08-3 as Open Item 15A-1 in the SER with open item.

In a letter 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 degrees C. The staff reviewed the changes and found them acceptable. RAI 15.08-3 is therefore resolved and Open Item 15.A-1 is closed. Verification that the applicant’s proposed changes are incorporated in the next revision of the FSAR is being tracked as **Confirmatory Item 15A-1**.

The cross-reference of Figure 15A-37 is changed from Figure 15A.6-32b to Figure 15A-38 and the word “Radiator” is changed to “Radiation.” These are editorial corrections and hence are acceptable.

The cross-reference of Figure 15A-38 is changed from Figure 15A.6-32a to Figure 15A-37. This is an editorial correction and hence is acceptable.

The cross-reference of Figure 15A-39 is changed from Figure 15A.6-33b to Figure 15A-40. This is an editorial correction and hence is acceptable.

The cross-reference of Figure 15A-40 is changed from Figure 15A.6-33a to Figure 15A-39. This is an editorial correction and hence is acceptable.

In Figures 15A-48, 51,52,53,64, 68, 69, changes are all editorial and hence acceptable.

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. This is a correction of the DCD and hence is acceptable.

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. This is a correction of the DCD and hence is acceptable.

The Figure 15A-70 cross-reference is changed from Tables 15A.6-2 through 15A.6-5 to Tables 15A-2 through 15A.6-5. This is an editorial change and hence is acceptable.

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 is acceptable.

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. These are editorial changes and hence are acceptable.

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. 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 “Plant Nuclear Safety Operational Analysis.” With the exception of **Confirmatory Item 15.A-1**, 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 plant NSOA that were incorporated by reference have been resolved.

However, as a result of the **Confirmatory Item 15.A-1**, the staff was unable to finalize the conclusions relating to “Plant Nuclear Safety Operational Analysis” in accordance with the NRC requirements.

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

15B.1 Introduction

This 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 incorporates by reference Appendix 15B of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in this appendix the applicant provides 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 and to Sections 15B.1 and 15B.4, and either deletes the name “Essential Multiplexing System“ or changes it to “Data Communication Function of the reactor trip and isolation system (RTIS) and the emergency safety features 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's regulations for the single-failure analysis, and the associated acceptance criteria, are in Section 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.

15B.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Appendix 15B of the certified ABWR DCD. The 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 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 staff reviewed the 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 COL application, Part 07, 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 COL application Part 07, Section 5.0 for a listing of all FSAR sections affected by this departure.

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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.

- 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 boiler system. This departure is 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. 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 FMEA that were incorporated by reference have been resolved.

15C Appendix C–Not used

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

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

Appendix 15D of the STP COL FSAR incorporates by reference, with no departures or supplements, Appendix 15D, “Probability Analysis of Pressure Regulator Downscale Failure,” of the certified ABWR DCD, Revision 4, which is referenced in 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 subsection. Pursuant to 10 CFR 52.63(a)(5) and 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 appendix is supplementary information supporting anticipated transient without scram (ATWS) performance evaluation.

15E.2 Summary of Application

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

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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.

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 provides 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 provides 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's regulations for the ATWS performance evaluation, and the associated acceptance criteria, are in Section 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, NRC staff reviewed and approved Appendix 15E of the certified ABWR DCD. The 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 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 staff reviewed the information in the COL FSAR:

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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.

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 COL application, Part 07, 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 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 has 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 Tier 1 departure STD DEP T1 3.4-1 and the Tier 2 administrative departure. However, no changes have been made to the ATWS logic-related text in the FSAR, specifically Sections 2.2.8 and 3.4 in FSAR Tier 1 and Section 7.4 and 15E.1 in FSAR Tier 2. No explanation of the logic changes shown on these revised FSAR figures has been provided in either of the cited departures. The staff also noted that some of the changes made to these FSAR figures are not annotated (back circled). Due to the above stated reasons, NRC staff was unable to verify the acceptability of the departed ATWS mitigation logic. Therefore, the staff issued RAI 15.08-2 requesting additional clarifications. The staff tracked RAI 15.08-2 as Open Item 15E-1 in the SER with open item.

In the response to the RAI 15.08-2 dated March 30, 2010 (ML100920022), the applicant updates FSAR Figure 15E-1b so that the ATWS mitigation logic now remains unchanged from the one described in the certified ABWR DCD. Changes to FSAR Figures 15E-1a, 15E-1b, and 15E-1c are now related to nomenclature changes identified in Tier 1 Departure STD DEP T1 3.4-1 and to some administrative editorial changes. The staff reviewed this RAI response and found 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 Tier 1 Departure T1 3.4-1 are evaluated in Chapter 7 of this SER. The RAI response is therefore acceptable. Open Item 15.E-1 is now considered closed. Revision 3 of the STP Units 3 and 4 COL FSAR does not capture the proposed FSAR changes. Verification of changes to FSAR Figure 15E-1b and corresponding changes to the ATWS technical specifications figures is being tracked as **Confirmatory Item 15.**E-1

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 provides a revised discussion for the automated initiation of the ADS inhibit. The 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 the 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.

- STD DEP Admin

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

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.

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. 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 “ATWS Performance Evaluation.” With the exception of **Confirmatory Item 15.E-1**, there is no outstanding information 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 probability analysis of a pressure regulator downscale failure that were incorporated by reference have been resolved.

However, as a result of the **Confirmatory Item 15.E-1**, the staff was unable to finalize the conclusions relating to “ATWS Performance Evaluation,” in accordance with the NRC requirements.

15F Appendix F – LOCA Inventory Curves

The staff’s evaluation of this appendix is included in Section 6.3 of this SER.