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 (AOO)

This COL license information item addresses 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 (DBAs)

This COL license information item addresses DBAs.

15.0.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed 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 Commission regulations for the review of transient and accident analyses, and the associated acceptance criteria, are given in Section 15 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

The regulatory basis for the review of COL license information items are given in Section 15 of NUREG-0800.

15.0.4 Technical Evaluation

As documented in NUREG-1503, “Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor,” the staff reviewed and approved Section 15.0 of the ABWR DCD. The applicant took no exceptions to Section 15.0 of the ABWR DCD. The staff reviewed Section 15 of the STP Units 3 and 4 COL FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1]

The staff’s review of this application includes COL License Information Items 15.1, 15.2, and 15.3.

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

- COL License Information Item 15.1 AOO

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 DBAs

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 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 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 letter U7-C-STP-NRC-090210 dated November 19, 2009. The

[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.

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 the certified DCD Chapter 15 includes the necessary analysis required for the core being licensed, the COL License Information items 15.0.5-1, 2 and 3 are satisfied and are considered closed.

15.0.5 Post Combined License Activities

There are no post COL activities related to this section.

15.0.6 Conclusion

The U.S. Nuclear Regulatory Commission (NRC) staff's finding related to information incorporated by reference is in NUREG–1503. The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the transient and accident analyses that were incorporated by reference have been resolved.

The staff has compared the application to the relevant regulations; acceptance criteria defined in NUREG-0800, Section 15.0, and other Regulatory Guides (RGs) and conclude that the applicant is in compliance with the regulations. COL License Information Items 15.1, 15.2, and 15.3 are adequately addressed by the applicant and are considered closed.

The staff's review confirmed that the applicant has addressed the relevant information, and there is no outstanding information expected to be addressed in the COL FSAR related to this section.

15.1 <u>Decrease in Reactor Coolant Temperature</u>

15.1.1 Introduction

This section of the FSAR addresses AOOs that increase the 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:

Commitments

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

Supplemental Information

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

15.1.3 Regulatory Basis

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

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

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

15.1.4 Technical Evaluation

As documented in Section 15.1 (1) of NUREG-1503, the staff reviewed and approved Section 15.1 of the ABWR DCD. The staff reviewed Section 15.1 of the STP Units 3 and 4 COL

FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1]

Commitments

In response to a COL Information Item in the 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 Rev. 3 of FSAR 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 letter U7-C-STP-NRC-090210 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

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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.

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 Section 5.1.2.3.2.2 includes the Feedwater Controller Failure-Maximum Demand analysis required for the core being licensed, the COL License Information Item 15.1-1 is satisfied and is considered closed.

Supplemental Information

The applicant submitted a new comparison Table 15.1S-2 listing the instrument response time given in the DCD and for STP Units 3 and 4. Response time was significantly changed from the DCD values assumed in the analysis for Scram, Safety Relief Function, Recirculation Pump Trip (RPT) and the Main Steam Isolation Valve (MSIV). Therefore, the staff issued (RAI 2694) RAI - 15.01.01-15.01.04-1 (STP Letter U7-C-STP-NRC-090061 dated July 2, 2009). The applicant stated that the changes were done inadvertently and the instrument delay times were returned to the values in the DCD and FSAR was revised in Revision 3 accordingly. Therefore, because the values were returned to the DCD values which were found acceptable in the FSER of 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 staff's finding related to information incorporated by reference is in NUREG-1503. The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the decrease in reactor coolant temperature that were incorporated by reference have been resolved.

The staff's review confirmed that the applicant has addressed the relevant information, and there is no outstanding information expected to be addressed in the COL FSAR related to this section.

15.2 Increase in Reactor Pressure

15.2.1 Introduction

This section of the FSAR addresses 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 air conditioning (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 MSIV automatic closure on 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.

Commitments

In FSAR Subsection 15.2.1.3.1, the applicant commits (COM 15.2-1) to provide an analysis of 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 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 MSIVs.

15.2.3 Regulatory Basis

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

In addition, the relevant requirements of the Commission regulations for the increase in reactor pressure, and the associated acceptance criteria, are given in Section 15.2.1-15.2.5 of NUREG-0800.

In addition, the relevant requirements for the Commission’s regulations and the associated acceptance criteria for the review of the COL license information items and supplemental information are in the 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 are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.A.4. Tier 2 departures affecting Technical Specifications (TS) 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 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 considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1]

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

The following Tier 1 departure identified by the applicant in this section require 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, please refer to COLA 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 require 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, please refer to COLA 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.

Commitments

In response to a COL action in the 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, 15.2-2).

"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."

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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.

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 letter U7-C-STP-NRC-090210 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. Because the certified DCD Sections 5.2.1.3.1, Inadvertent Closure of One Turbine Control Valve, 15.2.2.3.2.3, Generator Load Rejection with Failure of all Bypass Valves, include the analysis required for the core being licensed, the COL License Information items 15.2.1.3.1 and 15.2.2.3.2.3 are satisfied and are considered closed.

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 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).

In Section 15.2.10.1, “Radiological Effects of MSIV Closures,” of the STP FSAR, Revision 3, the applicant provided its radiological consequences of the inadvertent MSIV closure as 9.9 E-5 mGy for thyroid dose and 2.9 E-3 mGy whole body dose. The staff determined that these 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 finds 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.

15.2.5 Post Combined License Activities

There are no post COL activities related to this section.

15.2.6 Conclusion

The staff’s finding related to information incorporated by reference is documented in NUREG-1503. The staff’s review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the increase in reactor pressure that were incorporated by reference have been resolved.

The staff has compared the application to the relevant regulations; acceptance criteria defined in NUREG-0800, Section 15.0, and other RGs and conclude that the applicant is in compliance with the regulations**.**

The staff’s review confirmed that the applicant has addressed the relevant information, and no outstanding information is expected to be addressed in the COL FSAR related to this section.

15.3 Decrease in Reactor Coolant System Flow Rate

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

- Reactor internal pump (RIP) trip
- Recirculation flow controller failure (decreasing flow)
- Pressure Regulator Down-Scale 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 AOOs that cause a decrease in reactor RCS flow rate have been resolved.

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Introduction

This section of the FSAR, addresses AOO and 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 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:

Administrative Departures

- STP DEP Admin

The applicant identified two administrative departures in Subsections 15.4.2.1, "Features of the ABWR Automatic Thermal Limit Monitoring System (ATLM)," and 15.4.5.2.1.3, "Identification of Operator Actions"

[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 Items

- COL License Information Item 15.5 Mislocated Fuel Bundle Accident

This COL license information item addresses the mislocated fuel bundle accident.

- COL License Information Item 15.6 Misoriented Fuel Bundle Accident

This COL license information item addresses the misoriented fuel bundle accident.

15.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed 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 given in Section 15.4 of NUREG-0800.

The relevant requirements for the Commission's regulations and the associated acceptance criteria for the review of COL license information items is in Section 15.4 of NUREG–0800.

15.4.4 Technical Evaluation

As documented in NUREG-1503, the staff reviewed and approved Section 15.4 of the ABWR DCD. The staff reviewed Section 15.4 of the STP Units 3 and 4 COL FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1]

The staff reviewed the information provided in the FSAR:

Administrative Departures

STD DEP Admin

The applicant identified two administrative departures in Sections 15.4.2.1, "Features of the ABWR Thermal Limit Monitoring System and 15.4.5.2.1.3, "Identification of Operator Actions" The first departure deleted Reference 15.4.1 which was also deleted in the DCD and hence is 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 is a correction of the DCD and hence is 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.

COL License Information Items

- COL License Information Item 15.5 Mislocated Fuel Bundle Accident

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 letter U7-C-STP-NRC-090210 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. Because the certified DCD Section 15. 4.7.4 includes the Mislocated Fuel Bundle Accident analysis required for the core being licensed, the COL License Information item 15.5 is satisfied and is considered closed.

- 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 Section 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 letter U7-C-STP-NRC-090210 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 the certified DCD Section 15..4.8.3 include the Misoriented Fuel Bundle Accident analysis required for the core being licensed, the COL License Information item15.6 is satisfied and is considered closed

In Revision 3 of the FSAR, reference to General Electric Standard Application for Reactor Fuel (GESTAR) Amendment is deleted and replaced by "NRC approved methods." This is acceptable because in Reference 4.4.8, item 4.4.12, GESTAR Licensing topical report NEDE-24011-P-A is kept as a reference.

15.4.5 Post Combined License Activities

There are no post COL activities related to this section.

15.4.6 Conclusion

The staff's finding related to information incorporated by reference is in NUREG-1503. The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the decrease in RCS flow rate that were incorporated by reference have been resolved.

The staff's review confirmed that the applicant has addressed the relevant information, and no outstanding information is expected to be addressed in the COL FSAR related to this section.

15.5 Increase in Reactor Coolant Inventory

15.5.1 Introduction

This section of the FSAR addresses AOOs that cause an increase in reactor coolant system inventory. Increased inventory can be caused by 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 AOOs that cause an increase in 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, using the site-specific atmospheric dispersion factors (χ/Qs) provided in the STP FSAR, Revision 3, Section 2.3S.4.2.1.1, "Offsite Dispersion Estimates"; Table 2.3S-24, "PAVAN Results for Exclusion Area Boundary"; Table 2.4S-25, "PAVAN Results for Low Population Zone (LPZ)"; and Table 2.3S-25, "ARCON96 χ/Q Values."

The assessments are based on information provided in: (1) the ABWR DCD, Revision 4, Chapter 15, Section 15.6, "Decrease in Reactor Coolant Inventory," and (2) STP FSAR, Revision 2, 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 provided in the ABWR DCD Tier 1, Table 5.0, "Site Parameters," and in the ABWR DCD Tier 2, Table 15.6-13:

- Failure of a small line carrying primary coolant outside containment (DCD Section 15.6.2)
- Steam system piping break outside containment (DCD Section 15.6.4)
- Loss of coolant accident – inside containment (DCD Section 15.6.5)
- Cleanup water line break - outside 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 provides an evaluation of Sections 15.6 and 15.7 of the STP Units 3 and 4 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.

15.6.2 Summary of Application

Section 15.6 and Section 15.7 of the STP Units 3 and 4 COL FSAR incorporate by reference Section 15.6 and Section 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:

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 non-seismic. The applicant provides supplemental information in the COL FSAR Subsection 15.7.3.1, "Identification of Causes and Frequency Classification," identifying the radwaste building capabilities in preventing liquid releases.

Tier 2 Departures not requiring prior approval

- STP DEP 11.3-1 Gaseous Waste Management System

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

COL License Information Items

- COL License Information Item 15.7 Iodine Removal Credit

The applicant provides information to address COL License Information Item 15.7 in the FSAR, Subsection 15.6.7.1. This COL license information item describes the design characteristics of the main steam lines, drain lines, and main condensers for removing radioactive iodine prior to releasing it into the environment.

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

The applicant provides information to address COL License Information Item 15.9 in the FSAR, Subsection 15.7.6.1. This COL license information item describes the radiological consequences of postulated radioactive release due to liquid radwaste tank failure.

In the STP FSAR, Part 5, "Emergency Plan," Section G.3, "Technical Support Center," the applicant 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 an emergency response. The ABWR DCD Table 13.3-1 provides the ABWR design considerations for TSC emergency planning requirements.

15.6.3 Regulatory Basis

The regulatory basis for the review 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 are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.A.4. Tier 2 departures 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 the review of COL license information items is in Section 15.0.3 of NUREG–0800.

The ABWR design was certified 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, 1977 and for Testing Reactors," of 10 CFR Part 100, "Reactor Site Criteria." In the 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, Section 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 provided in Section 2.3.4 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 the 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

NRC staff reviewed Sections 15.6 and 15.7 of the STP Units 3 and 4 COL FSAR, Revision 3. The staff reviewed Section 15.6 of the STP Units 3 and 4 COL FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[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.

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 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-1503, the staff concluded that the bounding χ/Qs proposed by GE for the exclusion area boundary, the low-population zone, and the control room of the ABWR plant, in conjunction with the engineered safety features 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 consequences 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 consequences analyses that are impacted by the site characteristics. The applicant provided and discussed the STP site-specific offsite χ/Q values in Section 2.3S.4.2.1.1, "Offsite Dispersion Estimates." In Section 2.3.4, "Short-Term diffusion Estimates," of this SER, the staff discusses the review and evaluation of the STP site-specific χ/Q values provided in Section 2.3.4 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 2, Table 15.6.5S-1, "Site-Specific χ/Q," the applicant compared the STP site-specific offsite (exclusion area boundary[EAB] and low population zone[LPZ]) and control room χ/Q values with those values provided in the ABWR DCD Tier 1, Table 5.0, "Site Parameters," and in the ABWR DCD Tier 2, Tables 15.6-13 and 15.6-14. The applicant demonstrated that all STP site-specific offsite χ/Q at all time intervals are bounded (smaller) by the ABWR DCD generic site χ/Q values and therefore the STP site has better dispersion characteristics than the ABWR reactor design requires.

The staff finds that the STP site-specific χ/Q values are less than the comparable ABWR DCD χ/Q values, and therefore, the STP site-specific total dose for each DBA is less than the ABWR DCD generic total dose for each DBA. Therefore, the dose reference values specified in 10 CFR Part 100.11(a) are satisfied because the radiological consequences 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 control room χ/Q values, the staff finds that it 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. The staff issued RAI 15.00.03-1 related to FSAR Section 15.6.5S, "Site-Specific Design Basis Accident Doses," requesting the applicant provide the control room radiological consequence analyses for the DBAs that are relevant to the Turbine Building releases and to demonstrate the STP control room still meets the dose acceptance criterion specified in GDC 19. **This is tracked as Open 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 require prior NRC approval and the full scope of its technical impact may be evaluated in other sections of this SER accordingly. For more information, please refer to COLA 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 11.2 of this SER.

Tier 2 Departures not requiring prior 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, please refer to COLA 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 the STP FSAR, Revision 2, the applicant addressed COL License Information Item 15.7,"Iodine Removal Credit," stating that:

> "The design characteristics of the main steamlines, drain lines, and the main condensers 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 removal 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 finds 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 provided 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," of the ABWR DCD provides radwaste system failure EAB doses as a function of referenced χ/Q values in the ABWR DCD. In its review of the ABWR DCD, the staff reviewed and accepted the dose results in the ABWR DCD, Table 15.7-7. The applicant used the site-specific χ/Q values to determine the resulting doses from Table 15.7-7. The staff performed an independent dose calculation and finds the applicant's determination is correct and therefore finds it acceptable because the resulting doses are less than small fraction of the dose reference value specified in 10 CFR 100.11.

In the STP COL Part 5, "Emergency Plan," Section G.3, "Technical Support Center," the applicant stated that the TSC has the radiological protection and monitoring equipment necessary to control radiation exposure to any person working in the TSC of levels below 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of an accident as specified in NUREG-0737, Supplement No. 1, "Clarification of TMI Action Plan Requirements." It further states that the TSC design complies with 10 CFR Part 50, Appendix E, Section IV.E.8, for the radiological habitability and meets the TS radiological dose acceptance criterion specified in NUREG-0696. The staff issued RAI 13.03-73 related to the STP FSAR, Part 5, "Emergency Plan," Section G.3, "Technical Support Center," requesting the applicant provide the TSC radiological consequence analyses for the DBAs to demonstrate it meets the dose acceptance criterion specified in NUREG-0737, Supplement No. 1. **This is tracked as Open 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. The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the increase in reactor coolant inventory that were incorporated by reference have been resolved.

The staff's review confirmed that the applicant has adequately addressed the COL license information items in accordance with Section 15.0.3 of NUREG–0800.

However, as a result of Open Items 15.6-1 and 15.6-2, the staff is unable to finalize its conclusions as stated below:

Satisfactory resolution of Open Item 15.6-1 is needed in order for the NRC staff to conclude that the applicant has provided sufficient information to satisfy 10 CFR 100.11 (a), and GDC 19.

Satisfactory resolution of Open Items 15.6-1 and 15.6-2 is needed in order for the NRC staff to conclude that the STP Units 3 and 4 TSCs are in compliance with the 10 CFR Part 50, Appendix E, Section IV.E.8, for the radiological habitability and meets the TSC radiological dose acceptance criterion specified in NUREG-0696 and NUREG-0737, Supplement No. 1.

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to "Decrease in Reactor Coolant Inventory" and "Radioactive Release from Subsystems and Components." With the exceptions of Open Items 15.6-1 and 15.6-2, there is no outstanding information expected to be addressed in the COL FSAR related to this section. As a result of these open items, the staff is unable to finalize its conclusions relating to "Decrease in Reactor Coolant Inventory" and "Radioactive Release from Subsystems and Components" 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 shutdown the reactor must be addressed.

Section15.8 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 ATWS have been resolved.

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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.

Staff Evaluation of Chapter 15 Appendices

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 Appendix A to 10 CFR Part 52. The following departures were identified by the applicant:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

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

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 rewording of text in Subsection 15A.6.2.3.11, “Control Rod Worth Control.”

- STD DEP Admin

This is an administrative departure pertaining to rewording of text in Subsection15A.6.3.1, “General.”

15A.3 Regulatory Basis

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

In addition, the relevant requirements of the Commission regulations for the plant nuclear safety operational analysis, and the associated acceptance criteria, are given in Section 15 of NUREG-0800.

15A.4 Technical Evaluation

As documented in NUREG–1503, the 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 considered the referenced ABWR DCD to ensure that the combination of the information in the

COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1]

In addition, the staff reviewed the following:

The following Tier 1 departure identified by the applicant in this section require 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, please refer to COLA Part 07, Section 5.0 for a listing of all FSAR sections affected by this Tier 1 departure.

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. Figure 15A-7- Flammability Control System blocks shown in the figure are deleted. Since there is no Flammability Control System in STP Units 3 and 4, in accordance with 10 CFR 50.44(c)(2), the staff finds the deletion of the system in the figure is acceptable.

Administrative Departures

- STD DEP Admin

Figure 15A-13- Cross-reference to Figure 15A-67 is changed to Figure 15A-64. This is an editorial correction and hence is acceptable.

Figure 15A-17- Figure 15A-17-Cross reference to Figure 15A.6-57 is changed to 15A-64. This is an editorial correction and hence is acceptable.

Figure 15A-19- Added “flow” to Event title and the cross reference changed from 15A-57 to 15A-64. This is an editorial correction and hence is acceptable.

Figure 15A-21- Changed “Nuclear Boiler System” to “Main Steam System)” and the cross reference from 15A-19 to 1A-24. This is an editorial correction and hence is acceptable.

Figures 15A-25, 27-Cross reference changed from 15A-57 to 15A-64. This is an editorial correction and hence is acceptable.

Figure 15A-29- Added “automatic” in addition to manual scram during suppression pool high temperature. Since there is an automatic scram during suppression pool high temperature, this addition is correct and acceptable. It also deleted “instrumentation “from the Nuclear Boiler System block. This is an editorial correction and hence is acceptable. This departure also changed the set point for automatic suppression pool cooling from 38 degrees C to 35 degrees C. The NRC issued RAI 15.08-3 requesting the applicant to provide justification for this change being characterized as administrative.

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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.

Therefore, the staff's acceptance of changes to the figure 15A-29 is contingent upon the satisfactory resolution of RAI 15.08-3 (Open Item 15.A-1).

Figure 15A-37 - Changed "Radiator" to "Radiation" and the cross reference changed 15A.6-32b to 15A-38. This is an editorial correction and hence is acceptable.

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

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

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

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

Figure 15A-63- Changed fine motion control rod drive (FMCRD) to standby liquid control system (SLCS). This change is acceptable since the applicable event is reactor shutdown without control rods. This is a correction of the DCD and hence acceptable.

Figure 15A-67 - Deleted high-pressure core flooder (HPCF) connected to Leak Detection and Isolation System. There is no leak detection system for HPCF. This is a correction of the DCD and hence is acceptable.

Figure 15A-70 - Cross reference to (Tables 15A.6-2 through 15A.6-5) changed to (Tables 15A-2 through 15A.6-5). This is an editorial change and hence is acceptable.

Subsection 15A.6.2.3.11 (Control Rod Worth Control) - The bases 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, deletion of the control rod drop accident analysis with the applicable rod withdrawal error event is acceptable.

Subsection 15A.6.3.1 (General) - The events which are applicable for moderate frequency incidents are revised to include events 23, 26, 27, 38-40, 44, 45, 48 and 49. This is an editorial change and hence is acceptable.

15A.5 Post Combined License Activities

There are no post COL activities related to this section.

15A.6 Conclusion

The staff's finding related to information incorporated by reference is in NUREG-1503. The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the plant nuclear safety operational analysis that were incorporated by reference have been resolved.

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to "Plant Nuclear Safety Operational Analysis." With the exceptions of Open Item 15.A-1, there is no outstanding information expected to be addressed in the COL FSAR related to this section. As a result of this open item, the staff is unable to finalize its conclusions relating to "Plant Nuclear Safety Operational Analysis" in accordance with the NRC requirements.

Appendix B Failure Modes and Effects Analysis

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. In addition, in this appendix the applicant provides the following:

Tier 1 Departure

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

This departure describes changes to Table 15B-3 of the 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 Reactor Pressure Vessel (RPV) Water Level Instrumentation

This departure describes revisions to Figure 15B-1 to indicate the Nuclear Boiler Instrument Line Fill System.

15B.3 Regulatory Basis

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

In addition, the relevant requirements of the Commission regulations for the single failure analysis, and the associated acceptance criteria, are given in Section 15 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 identified a Tier 2 departure that does not require prior NRC approval. This departure is subject to the requirements of Section VIII, which are similar to the requirements in 10 CFR 50.59.

15B.4 Technical Evaluation

As documented in NUREG–1503, the staff reviewed and approved Section 15B of the certified ABWR DCD. The staff reviewed Appendix 15B of the STP Units 3 and 4 COL FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1]

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

The following Tier 1 departure identified by the applicant in this section require 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, please refer to COLA Part 07, Section 5.0 for a listing of all FSAR sections affected by this Tier 1 departure.

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

In Section 15 B.1 STD DEP T1 3.4-1 is provided, which refers to Table 15B-3. This change is in the Data Communication Function of the RTIS and the ELCS and the evaluation is addressed in Chapter 7. Failure Modes and Effects Analysis (FMEA) and common-cause failure (CCF) of RTIS and ELCS data communication functions is evaluated in Chapter 19.

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, please refer to COLA Part 07, Section 5.0 for a listing of all FSAR sections affected by this departure.

- STD DEP 7.7-1 RPV Water Level Instrumentation

In Subsection 15B.2.3 STD DEP 7.7-1 is provided which refers to 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 staff's finding related to information incorporated by reference is in NUREG-1503. The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to

[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.

10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the single failure analysis that were incorporated by reference have been resolved.

The staff's review confirmed that the applicant has addressed the relevant information, and no outstanding information is expected to be addressed in the COL FSAR related to this section.

Appendix C-Not used

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. 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.

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:

Tier 1 Departure

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

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

Administrative Departures

- STP DEP Admin

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

- STP 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 addressed in NUREG-1503.

In addition, the relevant requirements of the Commission regulations for the ATWS performance evaluation, and the associated acceptance criteria, are given in Section 15 of NUREG-0800.

15E.4 Technical Evaluation

As documented in NUREG–1503, the staff reviewed and approved Section 15B 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 reviewed the information in the COL FSAR:

Tier 1 Departure

The following Tier 1 departure identified by the applicant in this section require prior NRC approval and the full scope of its technical impact may be evaluated in other sections of this SER accordingly. For more information, please refer to COLA Part 07, Section 5.0 for a listing of all FSAR sections affected by this Tier 1 departure.

- STP 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:

STPNOC has revised the logic in the STP 3 & 4 FSAR Figures 15E-1a, “ATWS Mitigation Logic (alternate rod insertion (ARI), FMCRD Run-In, RPT, Manual Initiation),” 15E-1b, "ATWS Mitigation Logic (SLCS Initiation, Feedwater Runback)," and 15E-1c, “ATWS Mitigation logic (SLCS Initiation, Feedwater Runback)” based on Tier 1 departure STD DEP T1 3.4-1 and STD DEP Admin. However, no changes have been made to the ATWS logic-related text contained in the FSAR, specifically subsections 2.2.8 and 3.4 in Tier 1, and subsection 7.4 and 15E.1 in Tier 2. Explanation of the logic changes shown on these revised FSAR figures has not been provided in either of the cited departures. It was also noted that some of the changes made to these FSAR figures are not annotated (back circled). Due to the stated reasons, NRC staff is unable to verify the acceptability of the departed ATWS mitigation logic. STPNOC is asked to provide an explanation for all marked and unmarked logic changes shown on the revised FSAR figures 15E-1a, 15E-1b, and 15E-1c.

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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.

Therefore, the staff's acceptance of changes to the figures in Appendix E is contingent upon the satisfactory resolution of RAI 15.08-2 (Open Item 15.E-1).

Administrative Departures

- STP DEP Admin

In FSAR Revision 2 Section 15E.4, "ATWS Logic and Setpoints," the applicant provides a revised discussion for the automated initiation of ADS inhibit. The staff's review identified the following inconsistency:

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.

In response to staff RAI, (STP Letter U7-C-STP-NRC-09006 dated July 2, 2009), the applicant confirmed 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 issue is closed.

- STP 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 deletion of "vessel pressure" is acceptable since during a turbine trip with bypass valves open, the reactor pressure will decrease due to the opening of the turbine bypass valve. 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 staff's finding related to information incorporated by reference is in NUREG-1503. The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the probability analysis of pressure regulator downscale failure that were incorporated by reference have been resolved.

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to "ATWS Performance Evaluation." With the exception of Open Item 15.E-1, there is no outstanding information expected to be addressed in the COL FSAR related to this section. As a result of this open item, the staff is unable to finalize its conclusions relating to “ATWS Performance Evaluation” in accordance with the NRC requirements.

Appendix F – LOCA Inventory Curves

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