19.0 RESPONSE TO SEVERE ACCIDENT POLICY STATEMENT

This chapter describes the South Texas Project (STP) Units 3 and 4 plant-specific probabilistic risk assessment (PRA) and severe accident evaluations and corresponding regulatory requirements. In accordance with Title 10 *Code of Federal Regulations* (10 CFR) 52.79(a)(46), a combined license (COL) application is required to contain a description of the plant-specific PRA and its results. In addition, 10 CFR 52.79(d)(1) specifies that if the COL application references a design certification (DC), then the plant-specific PRA information must use the PRA information for the DC and be updated to account for site-specific design information and any design changes or departures.

19.1 Purpose and Summary (Related to RG 1.206, Part I, C.I.19, Appendix A, Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation," and Section 19.1,"Probabilistic Risk Assessment.")

19.1.1 Introduction

This section of the Final Safety Analysis Report (FSAR) described the text changes in Section 19.1 of the U. S. Advanced Boiling-Water Reactor (ABWR) design control documents (DCD) due to the departures of the South Texas Projects Unit 3 and 4 design from that described in the ABWR DCD. The applicant states that the consequence of these changes does not change the conclusion of the PRA in the ABWR DCD.

19.1.2 Summary of Application

Section 19.1 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.1 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in COL FSAR Section 19.1, the applicant provides the following:

Tier 1 Departures

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

This departure eliminates obsolete data communication technology and unnecessary inadvertent actuation prevention logic and equipment in the safety-related instrumentation and control (I&C) architecture. The departure also changes the implementation, architecture, testing, and surveillance descriptions for the Safety System Logic and Control (SSLC).

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 10.4-5 Condensate and Feedwater System

This departure states that the condensate booster pumps are part of the modified condensate and feedwater (FW) system.

19.1.3 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 Advance Boiling Water Reactor Design," (July 1994), (Final Safety Evaluation Report [FSER] related to the ABWR DCD).

In addition, in accordance with Section VIII, “Process 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, Tier 2*, and Tier 2 departures. Tier 1 departures require prior U.S. Nuclear Regulatory Commission (NRC) approval and are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.A.4. Tier 2* Departures require prior NRC approval and are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.6. Tier 2 departures affecting Technical Specifications require prior NRC approval and are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.C.4. Tier 2 departures that do not require prior NRC approval are subject to the requirements of 10 CFR Part 52, Appendix A, Section VIII.B.5, which are similar to the requirements in 10 CFR 50.59.

The regulatory basis for accepting the supplementary information relating to site-specific and plant-specific details and design features is established as follows:

- 10 CFR 52.79(a)(46), a description of the plant-specific probabilistic risk assessment (PRA) and its results.”

- 10 CFR 52.79(d)(1), which requires a COL applicant referencing a certified design (1) to include in the FSAR sufficient information demonstrating that the site characteristics fall within the site parameters specified in the DC; and (2) to have a plant-specific PRA information that must uses the PRA information from the DC, and is updated to account for site-specific design information and any design changes or departures.

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition,” (SRP), Section 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” Revision 2.

Regulatory Guide (RG) 1.206, Chapter C.I.19 also provides guidance for COL applicants, and C.III.19 provides guidance for a COL applicant referencing a certified design.

In addition, 10 CFR 52.79(a)(17) states that a COL application must contain an FSAR that provides the information with respect to compliance with technically relevant positions of the Three Mile Island (TMI) requirements in 10 CFR 50.34(f) of this chapter, with the exception of 10 CFR 50.34(f)(1)(xii), 10 CFR 50.34(f)(2)(ix), and 10 CFR 50.34(f)(3)(v).

10 CFR 52.79(a)(38) states that a COL application for a Light-Water Reactor (LWR) design must contain an FSAR that includes a description and analysis of design features for the prevention and mitigation of severe accidents, for example, challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.

The Staff Requirements Memorandum (SRM) dated July 21, 1993 on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs,” provides direction about severe accidents and the treatment of external events in PRAs to support DC and COL applications.

DC/COL-ISG-03, “Interim Staff Guidance, Probabilistic Risk Assessment Information to Support Design Certification and Combined License Applications,” dated June 11, 2008 (ML081430087), supplements the guidance provided to the staff in SRP Section 19.0 concerning the review of PRA information and severe accident assessment submitted to support DC and COL applications.

The regulatory requirement and guidance described in this section will be applicable to all subsequent sections in Chapter 19.

19.1.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 19.1 of the certified ABWR DCD. The staff reviewed Section 19.1 of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Purpose and Summary."

The staff reviewed the information in the COL FSAR:

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

The Tier 1 Departures identified by the applicant in this chapter requires prior NRC approval in the form of an exemption and the full scope of their technical impact may be evaluated in the other sections (or chapters) of this safety evaluation report (SER) accordingly. For more information, please refer to COL application Part 07, Section 5.0 for a listing of all FSAR sections affected by Tier 1 departures. In addition, compliance with 10 CFR Part 52, Appendix A, Section VIII.A.4 for Tier 1 departures will be addressed by the staff in a future exemption evaluation contained in Chapter 1 of this SER.

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

This departure eliminates obsolete data communication technology and unnecessary inadvertent actuation prevention logic and equipment. The departure also changes the implementation, architecture, testing, and surveillance descriptions for the SSLC. This departure states that a delta-PRA assessment was performed to determine the effect of the updates on the instrument and control fault trees (Appendix 19D) and on the common cause failures (CCFs), (Appendix 19N) of the essential communication function (ECF), as presented in Appendix 19D. However, these changes are not included in STP Units 3 and 4 FSAR Appendices 19D and 19N. NRC staff issued RAI 19.01-15 requesting the applicant to describe these changes and explain their impact on the PRA results.

The applicant's response to RAI 19.01-15 dated August 5, 2009 (ML092220163) states that the changes described in STD DEP T1 3.4-1 were evaluated using the plant-specific PRA model and no quantitative impact was determined, given the model described in the DCD and the design described in Departure STD DEP T1 3.4-1. The applicant also states that Table 19.2-2 of the STP COL FSAR will be revised to address the COL application changes noted in the RAI response. The staff found this response to RAI 19.01-15 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. Therefore, the response is acceptable and this RAI is resolved. Verification that the proposed revision is in the next revision of the COL application was tracked as Confirmatory Item 19-3 in the SER with open items. The staff confirmed that the proposed revision is incorporated into Chapter 19 of FSAR Revision 4, therefore, Confirmatory Item 19-3 is closed.

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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 2 Departure Not Requiring Prior NRC Approval

The Tier 2 Departures not requiring prior NRC approval identified by the applicant in this chapter may also be evaluated in other sections of this SER. For more information, please refer to COL application Part 07, Section 5.0 for a listing of all FSAR sections affected by these departures. In addition, the applicant's process for evaluating departures from the DCD is subject to NRC inspections. Finally, because 10 CFR 52.79(d)(1) requires the applicant to update the design certification PRA information to account for departures from the ABWR DCD, this Chapter of the SER also addresses how the plant-specific PRA has been updated to account for departures from the DCD, including departures not requiring NRC approval.

- STD DEP 10.4-5 Condensate and Feedwater System

The STP Units 3 and 4 design modification has four variable speed (adjustable speed drive [ASD]) Reactor FW Pumps and four condensate booster pumps. The original ABWR DCD design has three motor driven (MD) Reactor FW operating at full power. This departure increases the number of reactor FW pumps from three to four in the condensate and the FW system design. The departure also adds four condensate booster pumps to the system.

NRC staff issued RAI 19.01-20 asking the applicant to discuss the impact of these changes on the PRA results. In the response to this RAI dated August 5, 2009 (ML092220163), the applicant states that the value cited for the FW unavailability (Q), 5E-02, is determined by assuming that 50 percent of the time, FW pumps will trip on high water level and failure to manually recover at least one pump train is estimated at 0.1. The applicant states that the number of FW pumps (three) in the standard ABWR design does not affect the derivation of unavailability (Q) failure likelihood. Increasing the number of FW pumps to four in the STP Units 3 and 4 design does not affect the derivation of unavailability (Q). The staff found this approach acceptable and this RAI is resolved.

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

The staff also evaluated the impact of this departure on the PRA results. In addition, the staff also reviewed other departures in the later sections and appendices of Chapter 19 of this SER. Verification that the impact of departures on the PRA results is incorporated into the next revision of the FSAR is tracked as Confirmatory Item 19-15 as discussed in Section 19E.4 of this SER.

19.1.5 Post Combined License Activities

There are no post COL activities related to this section.

19.1.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information relating to "Purpose and Summary," 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 "Purpose and Summary" that were incorporated by reference have been resolved.

In addition, based on the above discussion on "Purpose and Summary," the staff concluded, pending the resolution of Confirmatory Item 19-15, that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.1S Additional Information to Support the COL Application

19.1S.1 Introduction

The applicant provides a cross-referenced table between the items in RG 1.206, Section C.I.19, Appendix A and the contents in the STP Units 3 and 4 FSAR.

19.1S.2 Summary of Application

Section 19.1S of the STP Units 3 and 4 COL FSAR provides supplemental information concerning the application in order to assist reviewers.

Supplemental Information

Table 19.1S-1 presents a cross-reference between the RG 1.206, Section C.I.19, Appendix A items and the format of the FSAR. Furthermore, the applicant assessed the risk significance of the PRA changes. The applicant states that the conclusions of the PRA are unaffected by any design change or site-specific analysis performed to support the COL application for the STP Units 3 and 4.

19.1S.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19.1S.4 Technical Evaluation

NRC staff reviewed Section 19.1S of the STP Units 3 and 4 COL FSAR. The staff reviewed the results of the STP Units 3 and 4 PRA and found them acceptable. Verification that the impact of departures on the PRA results is incorporated into the next revision of the FSAR is tracked as Confirmatory Item 19-15 as discussed in Section 19E.4 of this SER.

19.1S.5 Post Combined License Activities

There are no post COL activities related to this section.

19.1S.6 Conclusion

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 "Additional Information to Support the COL Application." No outstanding information is expected to be addressed in the COL FSAR related to this section. In addition, based on the above discussion on "Additional Information to Support the COL Application," the staff concluded, pending the resolution of Confirmatory Item 19-15, that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.2 Introduction (Related to RG 1.206, Part I, C.I.19, Appendix A, Section 19.1, "Probabilistic Risk Assessment"; Subsection 19.1.2.2, "PRA Level of Detail"; Subsection 19.1.4.1.1, "Description of the Level 1 PRA for Operation at Power"; Subsection 19.1.4.1.2, "Results from the Level 1 PRA for Operations at Power"; and Section 19.2.1, "Introduction.")

19.2.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.2 of the U.S. ABWR DCD due to the site-specific evaluations of the STP Units 3 and 4. The applicant states that the consequence of these changes does not change the conclusion of the PRA in the ABWR DCD.

19.2.2 Summary of Application

Section 19.2 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.2 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Section 19.2, the applicant provides the following:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP Admin (Table 19.2-1)

This departure corrects the referencing of key PRA assumptions on the reactor service water (RWS) system from FSAR Section 19.9.21 to Section 19.9.26.

Supplemental Information

Section 19.2.2 Objective and Scope

Table 19.2-2 in this section summarizes the effects of all listed departures in the COL FSAR on the PRA analysis and results.

Subsection 19.2.3.1 Key Assumptions and Ground Rules

The applicant updates the assumptions using supplemental site-specific information.

Subsection 19.2.3.2 Failure Probability and Field Experience

The applicant supplements the expected loss of offsite power (LOOP) frequency to reflect updated information and site-specific data used to calculate the PRA output.

Subsection 19.2.3.3 Initiating Accident Events

The expected LOOP frequency is supplemented to reflect updated information and site-specific data utilized to calculate the PRA output.

Subsection 19.2.4.4 External Consequence Analysis

The applicant updates the evaluation of external consequences with site-specific information using the MACCS computer code.

Subsection 19.2.4.5 Consequence Analysis Results

Using site-specific information, the applicant conducts evaluations and assesses them against the original results in Appendix 19E.3.

19.2.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19.2.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 19.2 of the certified ABWR DCD. The staff reviewed Section 19.2 of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Introduction."

The staff reviewed the information in the COL FSAR:

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

- STD DEP Admin (Table 19.2-1)

The applicant defines administrative departures as minor corrections, such as editorial or administrative errors in the referenced ABWR DCD (i.e., misspellings, incorrect references, table headings, etc.). The applicant identifies that this departure moves the COL action item, "Reactor Service Water System," from FSAR Subsection 19.9.21 to Subsection 19.9.26. The departure is only an administrative change and is therefore acceptable.

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

<u>*Supplemental Information*</u>

Section 19.2.2 Objective and Scope

Table 19.2-2 lists changes identified as DCD changes or revised structure, system, and component (SSC) design definitions. The table identifies those designs that can potentially

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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.

impact the PRA and the extent of the impact. NRC staff asked the applicant to provide additional information on the rationale for determining the impact of the departures on the PRA results.

The staff issued RAI 19.01-17 asking the applicant to discuss the impact of added components on the results of the interfacing systems loss-of-coolant accident (ISLOCA) analysis. In response to this RAI dated August 5, 2009 (ML092220163), the applicant states that in the ABWR DCD design all piping systems, major system components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) that extended outside the primary containment boundary are designed, to the extent practicable, to an ultimate rupture strength (URS) at least equal to full RCPB pressure. Accordingly, the ABWR DCD PRA does not include the ISLOCA as an initiating event. Therefore, upgrading the list of ISLOCA components does not change the assumption of an ISLOCA event in the PRA model. The staff found this response acceptable, and this RAI is resolved.

The staff issued RAI 19.01-13 requesting the applicant to discuss the impact of tripping condensate pumps in the event of a FW line break on the results of the PRA analysis. This RAI was tracked as open item 19-1 in the SER with open items. In the response to this RAI dated March 16, 2010, (ML100770391), the applicant states that the containment response to an FW line break inside the containment—without taking the credit for the automated condensate pump trip—is within the acceptable range. Therefore, taking this credit will have no negative impact on the PRA results. The staff agreed with this assessment, and this RAI is resolved.

The staff issued RAI 19-7 requesting the applicant to explain whether the manual switchover from one unit to the other unit for the fire protection system is modeled. If so, the RAI asked the applicant to describe the impact on the core damage frequency (CDF) from a fire event, as well as the impact of this single-fire protection system for the two units on the PRA results from an initiating event that can simultaneously affect both units (i.e., LOOP).

The applicant’s response to RAI 19-7 dated December 3, 2009 (ML093421266), states that Table 19.2-2 of STP FSAR Tier 2 will be revised to indicate that there is no significant effect on CDF, no change to the PRA, and only editorial changes to the fire protection system. The staff found that the applicant’s response is sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. Verification that the proposed revision is incorporated into the next revision of the FSAR was tracked as **Confirmatory Item 19-1** in the SER with open items. The staff confirmed that the proposed revision is incorporated into Chapter 19 of FSAR Revision 4, therefore, RAI 19-7 is resolved.

The staff issued RAI 19-8 requesting the applicant to clarify the residual heat removal (RHR) system heat removal rate and to explain whether the PRA results are impacted by this change in the RHR heat exchanger heat removal capacity.

The applicant’s response to RAI 19-8 dated July 13, 2009 (ML092740559), states that Table 19.2-2 of STP FSAR Tier 2 will be revised to indicate that the RHR heat removal rate increases to 0.427 megawatt per degree centigrade (MW/ºC). The staff found the response to RAI 19-8 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. The staff confirmed that the proposed revision is incorporated into Chapter 19 of the FSAR Revision 4, and RAI 19-8 is therefore resolved.

The staff issued RAI 19-9, requesting the applicant to explain whether key lock switches that are replaced with normal manual push-button switches are modeled in the PRA. If so, the RAI asked the applicant to describe the impact on the PRA results and the potential beneficial effect for plant-specific PRA.

The applicant's response to RAI 19-9 dated July 13, 2009 (ML092740559), states that the PRA was developed to support the DCD and is not extended to the level of detail to distinguish between key lock switches and push-button switches. Also, generic operator probabilities were used in the PRA, and the applicant states that operator response time to implement procedures is improved by the rotate and depress push-button action, instead of the slower response resulting from the administrative controls necessary when using key lock switches. The staff found the applicant's response to RAI 19-9 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19, and RAI 19-9 is therefore resolved.

The staff issued RAI 19-10 requesting the applicant to specify whether the described changes to the engineered safety features (ESF) Logic and Control System (ELCS) Mode are a clarification to the text or a design change and if the change is a design change, to explain to the staff how the PRA results are affected.

The applicant's response to RAI 19-10 dated July 13, 2009 (ML092740559) states that because there is a significant amount of time available for the operator to actuate the RHR system in the suppression pool cooling (SPC) after the core cooling function is successful, the operator action is modeled with a very low human error probability value of 6.5E-5/demand. The manual switch itself was not modeled. The switch design in the departure requires a certain "permissive" function be performed before the operator can initiate the RHR in the SPC mode, and the applicant characterizes this change as having the beneficial effect of reducing operator error. The applicant also states that the changes to the ELCS logic assure that the high-pressure core flooder (HPCF) "C" diverse hard-wired manual initiation function has priority over the normal automatic initiation logic for HPCF "C." This level of detail is not modeled in the PRA developed to support the DCD but is consistent with the intent of the PRA model. So the change does not have any impact on the PRA results. The staff found that this response to RAI 19-10 is sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. RAI 19-10 is therefore resolved.

The staff issued RAI 19-11 requesting the applicant to specify whether the described changes to the Containment Spray Logic Change are a clarification to the text or a design change. RAI 19-11 also requested the applicant to explain to the staff how the PRA results are affected if the change is a design change.

The applicant's response to RAI 19-11 dated December 3, 2009 (ML093421266) states that the departure clarifies the STP Units 3 and 4 containment spray logic design by (1) emphasizing that the low-pressure flooder (LPFL) mode has precedence over the containment spray below reactor vessel water Level 1, (2) clarifying the initiation of drywell and wetwell sprays, and (3) clarifying the interlocks associated with the RHR operation and clarifying that logic changes for the wetwell spray valves and suppression pool return valves do not change the DCD-required functional or safety requirements. The applicant also states that (a) the containment spray function is modeled in the internal events PRA prepared to support the DCD; (b) credit was taken for the containment spray function in evaluating the radioactive release consequences (categories and their frequencies); and (c) in this evaluation, the spray function is modeled with an operator action, but the control and logic associated with the spray function are not modeled and therefore, this departure will not change the PRA results. The staff found this response to RAI 19-11 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19, and RAI 19-11 is therefore resolved.

The staff issued RAI 19-12 requesting the applicant to specify whether the described changes to the RHR SPC modification are a clarification of the text or a design change. In addition, the RAI requested the applicant to explain to the staff how the PRA results are affected.

The applicant's response to RAI 19-12 dated July 13, 2009 (ML092740559), states that the departure clarifies the STP Units 3 and 4 RHR SPC logic design to provide (1) a more complete description of the SPC mode automatic and manual operations, and (2) more detail regarding the mode switch and its operation and to indicate that there are no changes in the DCD-required functional or safety requirements. The applicant also states that in the PRA that was prepared to support the DCD, the SPC mode is modeled as being initiated by an operator action. Also, the PRA does not model the details of the switch or the logic associated with the SPC mode of operation; they have a negligible impact on the PRA results compared to the operator action associated with the SPC mode of operation. Therefore, this departure has no impact on the results of the PRA. The staff found this response to RAI 19-12 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. Hence, RAI 19-12 is resolved.

The staff issued RAI 19-13 requesting the applicant to explain whether the safety relief valve solenoid valves are modeled in the PRA. If so, the RAI asked the applicant to describe the impacts of these changes on the PRA results, as well as the potentially beneficial effects on the plant-specific PRA.

The applicant's response to RAI 19-13 dated December 3, 2009 (ML093421266) states that the safety/relief solenoid valves are included in the PRA described by the DCD. However, the testing of the safety/relief valves (SRVs) described in Subsection 7.3.1.1.1.2(g) and modified by Departure STD DEP 7.3-16 is not included in the PRA described by the DCD. Because the DCD testing restriction states that the pilot solenoid valves can only be tested when the reactor is not pressurized (e.g., shutdown), there is no change to the PRA described in the ABWR DCD. The applicant also states that this departure removes the reactor pressure restriction, which allows testing to be performed at any pressure. The improved testing capabilities enhance the ability to schedule and perform planned and preventative maintenance, which leads to improved equipment reliability and reduces online unavailability. This improved equipment reliability is the potential benefit for the plant-specific PRA required to support plant operation in accordance with 10 CFR 50.71(h), which is identified in Table 19.2-2. The staff found this RAI response sufficient to meet the guidance in RG 1.206 and SRP Chapter 19, and RAI 19-13 is therefore resolved.

The staff issued RAI 19-14 requesting the applicant to specify whether the described changes to the reactor building cooling water system are a clarification to the text or a design change. RAI 19-11 also requested the applicant to explain to the staff how the PRA results are affected if the change is a design change.

The applicant's response to RAI 19-14 dated July 13, 2009 (ML092740559) states that Table 19.2-2 of the STP FSAR Tier 2 will be revised to remove the statement "clarification to text" and no "direct" effect on PRA. This engineering change supports an increased heat removal capacity and corrects inconsistencies in Subsection 9.2.11.2. The staff found this response to RAI 19-14 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. The staff verified that the proposed revision was incorporated into Revision 4 of the FSAR. In a supplemental response to RAI 19-14 dated January 20, 2010 (ML100250138), the applicant explains the screening process for developing the plant-specific PRA model in accordance with the guidance described in RG 1.206, Section C.III.I.19. Based on this process, eleven changes or departures remained after the preliminary screening. In addition, the site-specific ultimate heat sink (UHS) design is included in the final evaluation process. The staff found this process acceptable for identifying the impact of the departures on the PRA model changes. Therefore, based on the above discussion, RAI 19-14 is resolved.

Subsection 19.2.3.1 Key Assumptions and Ground Rules

The applicant did not provide in FSAR Subsection 19.2.3.1, Revision 2, the supplemental information relating to the key assumptions. The staff issued RAI 19.01-22 asking the applicant to provide this information. The applicant's response to RAI 19.01-22 is discussed in detail in Section 19.3.4 of this SER. Based on this response, the assumptions of the STP COL application PRA have been supplemented with updates from site-specific information. Based on the staff's review of Section 19.3.4, this RAI is resolved.

Subsection 19.2.3.2 Failure Probability and Field Experience

See Section 19.3. Section 19.3 reviews the updated LOOP frequency.

Subsection 19.2.3.3 Initiating Accident Events

There is only one change in the initiating accident event frequency (i.e., LOOP frequency). Section 19.3 addresses the results of the review.

Subsection 19.2.4.4 External Consequence Analysis

The applicant updates the evaluation of external consequences with site-specific information using the MACCS2 computer code. The complete review and the results of the review are in Section 19E.4 of this SER.

Subsection 19.2.4.5 Consequence Analysis Results

The evaluations and reviews of the results are in Section 19E.4 of this SER.

19.2.5 Post Combined License Activities

There are no post COL activities related to this section.

19.2.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information relating to "Introduction." 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 "Introduction" that were incorporated by reference have been resolved. In addition, based on the above discussion on "Introduction," the staff concluded that the relevant information in the COL FSAR, Revision 4 is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.3 Internal Event Analysis (Related to RG 1.206, Part I, C.I.19, Appendix A, Subsections 19.1.4.1.1, "Description of the Level 1 PRA for Operation at Power"; 19.1.4.1.2, "Results from the Level 1 PRA for Operations at Power"; 19.1.4.2.1, "Description of the Level 2 PRA for Operations at Power"; 19.1.4.3.1, "Description of the Level 3 PRA for Operations at Power"; and Section 19.2, "Severe Accident Evaluation.")

19.3.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.3 of the U.S. ABWR DCD due to the site-specific changes of the STP Units 3 and 4. The applicant states that the PRA results and insights are still in compliance with the conclusion of the PRA in the ABWR DCD.

19.3.2 Summary of Application

Section 19.3 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.3 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Section 19.3, the applicant provides the following:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure addresses the issue that the reactor core isolation cooling (RCIC) pump and turbine are contained in the same casing on a monoblock. The design eliminates many supporting components.

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

This departure eliminates obsolete data communication technology and unnecessary inadvertent actuation prevention logic and equipment. The departure also changes the implementation, architecture, testing, and surveillance descriptions for the SSLC.

- STP DEP T1 5.0-1 Site parameters

This departure addresses information pertaining to STP Units 3 and 4 site parameters. The information which is not bounded by the ABWR DCD is described in the FSAR.

Tier 2 Departure Requiring Prior NRC Approval

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

This design change utilizes two medium voltage electrical systems (MVES) (13.8 kilovolt [kV] and 4.16 kV) instead of the one 6.9 kV electrical system described in the ABWR DCD.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 2.2-5 CRAC2 and MACCS2 Code

This departure replaces the CRAC2 code with the MACCS2 computer code; the CRAC2 code was used in the DCD.

- STP DEP 9.2-5 Reactor Service Water (RSW) System

This departure increases the reactor service water (RSW) flow rate required for the increased heat load from the STP Units 3 and 4 designs.

- STD DEP 10.4-5 Condensate and Feedwater System Design (Table 19.3-2)

This departure increases the number of reactor feed pumps from three to four and adds four condensate booster pumps to the system.

- STD DEP 19.3-1 Evaluation of Common Cause Failures

This departure addresses common cause failure (CCF) factors identified in the ABWR DCD review process and added to the STP Units 3 and 4 PRA model.

- STD DEP Admin

This departure addresses corrections in the cross-references of various sections in the ABWR DCD.

Supplemental Information

Section 19.3.1 Frequency of Core Damage

The applicant reviews the impact of these departures on the site-specific PRA results.

Subsection 19.3.1.1 Accident Initiators

The applicant describes the evaluation to verify that the overall risk impact of grid events at STP Units 3 and 4 is bounded by the original SSAR Appendix 19D analysis.

Subsection 19.3.1.3 Accident Sequence Analysis

The applicant uses the modified condensate and FW system as a frontline system in the PRA analysis.

Subsection 19.3.1.4 Frequency of Core Damage

The applicant evaluates the impact of the above departures on the frequency of core damage.

Subsection 19.3.1.5 Results in Perspective

The applicant discusses the qualitative results of a Level 1 internal event at power in the context of the above departures.

Section 19.3.3 Magnitude and Timing of Radioactive Release

The applicant changes the location of the results; these are administrative changes.

Subsection 19.3.4 Consequence of Radioactive Release

The applicant states that the MACCS2 computer code was used to calculate the consequences of potential radioactive releases.

19.3.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19.3.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 19.3 of the certified ABWR DCD. The staff reviewed Section 19.3 of the STP Units 3 and 4 COL FSAR, and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Internal Event Analysis."

The staff reviewed the information in the COL FSAR:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump

The pump and turbine are contained in the same casing on a monoblock. This design eliminates many supporting components. NRC staff issued RAI 19.01-14 asking the applicant to describe how the new design is modeled in the STP Units 3 and 4 plant-specific PRA model. The applicant states that the lubrication system basic event and other supporting component basic events, including the condensate pump, the barometric condenser, and the vacuum pump, were removed from the PRA model because these components were eliminated from the new design. The results show that the impact on the CDF is minimal. The staff performed an audit (ML093560778) on the RCIC model changes and confirmed that the impact of the RCIC change on CDF is minimal. Therefore, this RAI is resolved.

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

This departure eliminates obsolete data communication technology and unnecessary inadvertent actuation prevention logic and equipment. The evaluation of this departure is in Section 19.1.4 of this SER.

- STP DEP T1 5.0-1 Site Parameters

STP Units 3 and 4 site parameters are not bounded by the site parameter descriptions in the ABWR DCD. Appendix 19R of this SER describes and evaluates the effect of this departure on the external flooding analysis.

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 is evaluated in the other sections of this SER

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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.

accordingly. For more information, refer to COL application Part 07, Section 5.0, for a listing of all FSAR sections affected by this departure.

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

The ABWR standard Reference Combined License (R-COL) design modification states that dual MVES consisting of 13.8 kV and 4.16 kV will replace the single 6.9 kV MVES in the ABWR DCD. NRC staff issued RAI 19.01-18 asking the applicant to provide a list of PRA components that are supported by the 13.8 kV and 4.16 kV systems. The applicant's response to this RAI dated August 5, 2009 (ML092220163), states that there is no change in divisional Class 1E bus loads and only minor shifts in the non-Class 1E bus loads, between 6.9 kV and the new 13.8/4.16 kV buses. However, the applicant does not provide the basis for how the new basic event failure rates are calculated. RAI 19.01-18 was tracked as open item 19-3 in the SER with open items. In a supplemental response dated December 3, 2009 (ML093421266), the applicant provides the basis for the new basic events. The staff's review indicated that there were no reported differences in failure data between different distribution voltage designs. Therefore, it is acceptable to use the data supporting the ABWR PRA for the revised 13.6KV/4.16KV distribution system for STP Units 3 & 4. Based on this discussion, RAI 19.01-18 is resolved.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 2.2-5 CRAC2 and MACCS2 Code

This departure replaces the CRAC2 computer code with the MACCS2 computer code. This evaluation of this departure is in Section 19E.4 of this SER.

- STP DEP 9.2-5 Reactor Service Water (RSW) System

This departure modifies the RSW and the UHS system designs to meet the increased heat removal requirements of the reactor cooling water (RCW) system for STP Units 3 and 4.

The applicant's evaluation of this departure described above, in accordance with Item B.5 of Section VIII, determined that this departure does not require prior NRC approval. The staff reviewed the Departures Report regarding this departure, and could not determine whether it is reasonable for this departure not to require prior NRC approval. Therefore, the staff issued RAI 19.01-19 asking the applicant to describe the changes in the STP Units 3 and 4 plant-specific PRA model and explain the impact the changes have on the PRA results. In the response to this RAI dated August 5, 2009 (ML092220163), the applicant states that the addition of the UHS cooling fans resulted in an approximate 10 percent increase in division failure frequency. A normally open motor-operated valve (MOV) was added to the RSW pump discharge with no significant effect on PRA results due to the low failure rate for a normally open valve. The overall CDF increase due to the RSW-UHS design is small. The staff performed an audit (ML093560778) and confirmed that the effect on CDF is small. Therefore, this RAI is resolved.

- STD DEP 10.4-5 Condensate and Feedwater System

This departure increases the number of reactor feed pumps from three to four in the condensate and FW system design. The evaluation of this departure is addressed in Section 19.1.4.

- STD DEP 19.3-1 Evaluation of Common Cause Failures

Based on Section 19D.8.6 of the ABWR Standard Safety Analysis Report (SSAR), the following SSCs are considered in the CCF sensitivity analysis for the HPCF, RHR, reactor building cooling water (RBCW), and reactor building service water (RBSW) systems: pumps, pump auxiliary equipment, manual valves, MOVs, check valves, room air conditioners, spargers, strainers, circuit breakers, flow transmitters, heat exchangers, and temperature elements. CCF factors identified in the ABWR SSAR were added in the STP Units 3 and 4 PRA model. However, related to RAI 19.01-22 and the audit of the STP Units 3 and 4 PRA conducted at the Nuclear Energy Institute (NEI) office in Rockville, Maryland, during September 22 and 23, 2009, CCF is modeled for the pumps of the RBSW and RBCW systems. It is not clear, however, whether CCFs are being considered for other systems and components (e.g., HPCF and RHR). The applicant subsequently revised its PRA model to include the HPCF and RHR systems; therefore RAI 19.01-22 is resolved.

- STD DEP Admin

The applicant defines administrative departures as minor corrections, such as editorial or administrative errors in the referenced ABWR DCD (i.e., misspellings, incorrect references, table headings, etc.). The applicant identifies corrections to the appropriate reference in Subsection 19.3.3, “Magnitude and Timing of Radioactive Release,” of the ABWR DCD. This change corrects the cross-referencing in the DCD and has no impact on the results presented in the DCD or the COL FSAR. NRC staff found this change acceptable.

The applicant’s evaluation in accordance with 10 CFR Part 52, Appendix A, Section VIII, item B.5 determined that the departures do not require prior NRC approval. The departures have been evaluated in other chapters of the SER and the NRC staff finds it reasonable that the departures do not require prior NRC approval.

NRC staff also evaluated the impact of the departures on the PRA results. The results of the evaluation are shown below.

<u>*Supplemental Information*</u>

Section 19.3.1 Frequency of Core Damage

The applicant reviewed the impact of the departures on the site-specific PRA results. The evaluation includes the departures described above, except STD DEP 2.2-5, which is evaluated in Appendix 19E.

Subsection 19.3.1.1 Accident Initiators

The applicant describes the evaluation verifying that the overall risk impact of grid events at STP Units 3 and 4 is bounded by the analysis in Subsection 19D of the referenced DCD. NRC staff issued RAI 19.01-1 requesting the applicant to describe the quantitative information used to determine that the risk impact of the LOOP events at STP Units 3 and 4 is bounded by the analysis in Subsection 19D of the referenced DCD.

The applicant’s response to RAI 19.01-1 dated July 23, 2009 (ML092080083), states that a sensitivity analysis comparing the ABWR LOOP results, including initiating event frequency and recovery data, to similar area specific data in NUREG/CR–6890 was performed for the STP Units 3 and 4 plant-specific PRA model and re-performed using the reconstituted PRA model of the ABWR. Using the data from NUREG/CR–6890 for the Energy Reliability Council of Texas (ERCOT), there is a decrease in CDF from the LOOP initiating events, which confirms that the frequency estimates for the LOOP events used in SSAR Subsection 19D.3.1.2.4, including specific causes such as a severe storm, are bounding for the STP Units 3 and 4 site. The STP FSAR will be revised to clarify the use of the NUREG/CR–6890 LOOP data and the results of the sensitivity analysis.

During the staff's audit of the STP Units 3 and 4 PRA in September 2009 (ML093560778), the staff reviewed the applicant's detailed quantitative calculation used to determine that the risk impact of LOOP events at STP is bounded by the analysis in Chapter 19D of the SSAR. This evaluation also addressed COL License Information Item 19.6 (see Section 19.9 of this SER for additional details). This detailed calculation included a sensitivity analysis comparing the LOOP PRA results of the SSAR, including LOOP frequency and recovery data, to similar area specific data using the ERCOT regional information in Table 3-6 of NUREG/CR–6890. The power recovery distribution for STP Units 3 and 4 is consistent with that used in the SSAR. The detailed calculation showed a decrease in CDF from LOOP-initiating events for STP Units 3 and 4, which confirms that the frequency estimates for the LOOP events used in SSAR Subsection 19D.3.1.2.4 are bounding for the STP Units 3 and 4. However, the staff determined that the applicant did not actually use the ERCOT regional LOOP frequency (i.e., 0.0262/reactor-critical-year). Instead, the applicant used the plant-level, industry average LOOP frequency in Table 3-1 of NUREG/CR–6890 (i.e., 0.0359/reactor-critical-year). This discrepancy, however, does not change the conclusion that the frequency estimates for the LOOP events used in SSAR Subsection 19D.3.1.2.4 are bounding for the STP Units 3 and 4. Based on the above observation, the applicant agrees to revise the detailed calculation using the ERCOT data and to resubmit the response to RAI 19.01-1.

The applicant’s revised response to RAI 19.01-1 dated December 3, 2009 (ML093421266) appropriately uses the ERCOT regional LOOP frequency. Based on the above discussion, the staff found that the applicant's response to RAI 19.01-1 sufficiently addresses the concerns associated with this RAI. Verification that the applicant’s proposed revisions are included in the next revision of the COL application was tracked as Confirmatory Item 19-4 in the SER with open items. The staff confirmed that the proposed revisions are incorporated into Chapter 19 of FSAR Revision 4, therefore, RAI 19.01-1 is resolved.

Subsection 19.3.1.3 Accident Sequence Analysis

The PRA analysis used the modified condensate and FW system as a front-line system. See the discussion under Departure STD DEP 10.4-5 in this section.

NRC staff conducted an audit of the STP Units 3 and 4 PRA, which supports Chapter 19 of the STP Units 3 and 4 FSAR. The audit was conducted at the NEI office in Rockville, Maryland, during September 22 and 23 of 2009. Before the audit, the staff reviewed the accident sequence analysis in the ABWR SSAR, including selected event trees in Section 19D of this report. The staff also reviewed Subsection 19.3.1.3 of the STP Units 3 and 4 FSAR Revision 2, for departures. Based on this review, staff chose the following two at-power internal event trees in the SSAR for comparison against the reconstituted STP CAFTA model (REC model) during the audit:

- Large break loss-of-coolant accident
- Inadvertent opening of relief valve

The REC model event trees were found to be functionally identical to those in the SSAR. No top events in the Level 1 event trees were found for the control rod drive (CRD) flow, the containment overpressure protection system (COPS), and the firewater addition system in either the SSAR or the REC models. The staff further verified that the CRD flow and firewater addition are not explicitly modeled in the pertinent STP fault trees.

Subsection 19.3.1.3.1 Success Criteria

NRC staff conducted an audit of the STP Units 3 and 4 PRA, which supports Chapter 19 of the STP Units 3 and 4 FSAR. The audit was conducted at the NEI office in Rockville, Maryland, during September 22 and 23 of 2009. Before the audit, the staff reviewed the success criteria described in Subsection 19.3.1.3.1 of the SSAR and tabulated in Table 19.3-2 of the SSAR. The staff also reviewed the changes to the success criteria described in Table 19.3-2 of the STP Units 3 and 4 FSAR. The only departure in the STP success criteria table requires the addition of a condensate booster pump wherever a condensate pump appears in the corresponding SSAR table.

The staff requested verification that the discharge pressure of the condensate booster pump would be sufficient to overcome reactor pressure vessel backpressure for the events of interest. The staff confirmed that the discharge pressure of the condensate booster pump is equivalent to that of the original condensate pump described in the SSAR and is adequate to provide injection, as specified in the success criteria of Table 19.3-2 of the STP Units 3 and 4 FSAR.

The staff issued RAI 19.01-30 requesting the applicant to confirm that no credit is taken for firewater addition to the reactor vessel in the calculation of the baseline CDF. In response to RAI 19.01-30, the applicant stated that firewater addition system pump could prevent initial core damage, but this capability was conservatively ignored in the PRA.

The applicant substantially reconstituted the Level 1 internal events PRA from the SSAR and made sequence-by-sequence comparisons between the REC model and the SSAR PRA. A number of significant discrepancies arose when no credit was taken for the CRD flow and the COPS (as well as RHR recovery actions before containment failure and core damage) in the Level 1 REC model. These differences can be substantially reconciled when credit for the CRD and COPS (and, apparently, RHR recovery) is taken via post-processing of the relevant accident sequence frequencies. Specifically, without credit for the CRD flow (or credit for recovery of some other high pressure injection system) in the REC model, a number of sequences can be as much as an estimated order of magnitude higher in frequency than the corresponding SSAR PRA results. When integrating overall sequences, credit for the CRD flow reduces CDF by about 3 percent. Likewise, credit for the COPS (and apparently, for RHR recovery) reduces the estimated internal CDF events by about a factor of 3 to 4. Although the CRD flow is not explicitly described as part of the success criteria in Table 19.3-2, the CRD flow (or recovery of some other high pressure injection system) may be credited for several events in the reconstituted PRA model. Therefore, the staff issued RAI 19.01-30 requesting the applicant to clarify the following statement in Subsection 19.3.1.3 of the STP Units 3 and 4 FSAR:

> The Control Rod Drive (CRD) pumps which have limited capacity have not been included in the success criteria.

The staff's review of the SSAR also identified that although credit for the COPS is not explicitly modeled in the Level 1 PRA event trees, credit can be found in the containment event trees. For example, Figure 19D.5-10 of the SSAR (Amendment 33) shows the containment event trees for the Class II plant damage state and corresponding sequences. The COPS rupture disk opening for the branch path with no RHR recovery leads to successful core cooling and no core damage. Thus, the staff issued RAI 19.01-30 requesting clarification regarding the extent to which credit is taken for the COPS for relevant events.

The applicant's response to RAI 19.01-30 dated November 3, 2009 (ML093140253), clarifies the success criteria and the extent to which a number of systems are credited in the Level 1 PRA for STP Units 3 and 4. These systems include the CRD flow, COPS, RHR recovery, and AC-independent water addition. The staff found that the applicant's response to RAI 19.01-30 (parts 1 through 3) clarifies how these systems are or are not credited in the PRA by identifying the appropriate sections and text in the DCD and SSAR and by the fact that these sections are "incorporated by reference" in the STP Units 3 and 4 FSAR. The staff considered the applicant's response to RAI 19.01-30 acceptable, and RAI 19.01-30 is resolved.

Subsection 19.3.1.4 Frequency of Core Damage

The applicant evaluated the impact of the departures on the CDF. The staff issued RAI 19.01-22 asking the applicant to provide the quantitative results and the discussions of those results. This RAI was tracked as open item 19-2 in the SER with open items. The applicant's response to this RAI dated August 5, 2009 (ML092220163), states that the cumulative impact of the STP plant-specific CDF is less than a 10 percent change in CDF relative to the design certification PRA. Therefore, there is no need to provide the quantitative results according to the interim staff guidance (ISG) described in DC/COL-ISG-03. The staff performed another audit on March 31, 2010 (ML110260193) and examined the results of the STP plant-specific PRA. The staff found the results acceptable, and this RAI is therefore resolved.

Subsection 19.3.1.5 Results in Perspective

The applicant provided the qualitative results of a Level 1 internal event at power in the context of the above departures. The staff issued RAI 19.01-22 asking the applicant to provide the quantitative results and the discussions of those results. This RAI was tracked as open item 19-2 in the SER with open items. Based on the conclusion of the above discussion in Subsection 19.3.1.4, this RAI is resolved.

Section 19.3.3 Magnitude and Timing of Radioactive Release

The applicant makes administrative changes to the location of the results. This discussion is described in Section 19E.4 of this SER.

Section 19.3.4 Consequence of Radioactive Release

The applicant states that the MACCS2 computer code was used to calculate the potential radioactive release. This discussion is described in Section 19E.4 of this SER.

19.3.5 Post Combined License Activities

The applicant identifies commitment (COM 19.9-2) to address COL License Information Item 19.2 as discussed in Section 19.9.4 of this SER.

19.3.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Internal Event Analysis." 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 "Internal Event Analysis" that were incorporated by reference have been resolved.

In addition, based on the above discussion on "Internal Event Analysis," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.4 External Event Analysis and Shutdown Risk Analysis (Related to RG 1.206, Part I, C.I.19, Appendix A, Section 19.1.5, "Safety Insights from the External Events PRA for Operations at Power," and Subsection 19.1.6.1, "Safety Insights from the PRA for Other Modes of Operation.")

19.4.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.4 of the U. S. ABWR DCD due to the site-specific changes of the STP Units 3 and 4. The applicant states that the PRA results are bounded by the conclusion of the ABWR DCD with the exception of the probabilistic flooding analysis. This site-specific analysis was performed, and the results are discussed in Section 19R.

19.4.2 Summary of Application

Section 19.4 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.4 of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Section 19.4, the applicant provides the following:

Tier 1 Departure

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

This departure addresses the determination that the radwaste building (RWB) is not classified as a Seismic Category I structure.

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP Admin

This departure addresses the proposed administrative departure from the ABWR DCD that entails minor corrections in the referenced ABWR DCD (e.g., misspellings, etc.).

Supplemental Information

Subsection 19.4.3.2.1 Structure Fragility

Because of the reclassification of the radwaste building from Seismic Category 1 to non-seismic in the Departure STD DEP T1 2.15-1, no seismic fragility for this building is evaluated.

Subsection 19.4.3.4 Results of the Analysis

The applicant states that the STP Units 3 and 4 site-specific geology is bounded by the ABWR DCD seismic design.

Section 19.4.4 Fire Protection Probabilistic Risk Assessment

The applicant reviews the impact of proposed plant departures on the results of the ABWR DCD Fire-Induced Vulnerability Evaluation (FIVE) analysis. The applicant concludes that the existing ABWR FIVE results bound the STP Units 3 and 4 fire analysis.

Section 19.4.5 ABWR Probabilistic Flooding Analysis

The applicant provides site-specific supplemental information that addresses the probabilistic flood analysis of the relocated RSW pump house and external flooding.

19.4.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19.4.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 19.4 of the certified ABWR DCD. NRC staff reviewed Section 19.4 of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "External Event Analysis and Shutdown Risk Analysis."

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

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

The referenced ABWR DCD Section 2.15.13 states that the exterior walls of the RWB below grade and the basemat are classified as Seismic Category I structure. This departure revises the seismic category of the RWB substructure from Seismic Category I to non-seismic. 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.

RWB does not house any safety-related systems or components. RG 1.29, “Seismic Design Classification,” provides a list of SSCs that have to be classified as Seismic Category I. Item “p” on page 4 of RG 1.29 states, “systems, other than radioactive waste management systems, not covered by …“ shall be Seismic Category I. The phrase "other than radioactive waste management systems” excludes these systems from the list of Seismic Category I SSCs. For the radioactive waste management system, RG 1.29 refers to RG 1.143 in Note 5. The detailed guidance for the design of the radwaste processing SSCs is in RG 1.143.

This departure commits to follow the guidance of RG 1.143. Also, NUREG–1503 Section 3.8.4 states that the RWB is not a Seismic Category I. The NRC staff’s review included this design because General Electric (GE) elected to design the RWB substructure as a Seismic Category I.

Based on this departure, the COL FSAR was revised to delete the description and results of the RWB analysis and design from those sections of the ABWR DCD, which included the description because the RWB substructure was classified as a Seismic Category I structure. Examples of these deleted sections include Sections 2.5S.4, 3.7, 3.8, and Appendix 3H.3. Also, revisions throughout the COL application have appropriately changed the seismic classification of the RWB (Part 7, Table 5.0-1).

The staff’s evaluation determined that there was a need for additional information before accepting Departure STD DEP T1 2.15-1. Specifically, the staff issued RAI 19-24 requesting the applicant to confirm that a failure of the RWB under seismic and tornado loadings will not impact the adjacent Seismic Category I buildings and equipment. The staff requested the applicant to state the physical separation of the RWB from Seismic Category I buildings.

In the response to RAI 19-24 dated August 26, 2009 (ML092430135), the applicant confirms that the RWB will be designed against collapse when exposed to seismic or tornado loadings. In accordance with the acceptance criteria in SRP Sections 3.3.1, 3.3.2, and 3.7.2 that allow for a design against the collapse of non-seismic Category I buildings onto Seismic Category I SSCs, the staff issued RAI 19-33 requesting the applicant to provide generic design procedures for SSCs with interaction potential to resist site-specific external events (e.g., wind, tornado, and seismic events). The staff needed this information to conclude with reasonable assurance that the applicant will adequately analyze and design the RWB against collapse when exposed to seismic or tornado loadings, in compliance with GDC 2 and relevant SRP acceptance criteria. This RAI was tracked as Open Item 19-14.

The applicant’s response to RAI 19-33 dated June 16, 2010 (ML101690148), refers to the response to RAI 03.08.04-18 Revision 1 dated June 2, 2010 (ML101580248), which provides the design procedures for the Seismic Category II/I design of the RWB for seismic and tornado loadings. The staff noted that RAI 19-33 addresses issues similar to those formulated in RAI 03.08.04-18, and referencing this RAI in the response is acceptable. Therefore, RAI 19-33 and Open Item 19-14 are considered resolved and closed. The staff evaluation of RAI 03.08.04-18 is discussed in SER Chapter 3 Section 3.8.

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP Admin

The applicant defines administrative departures as minor corrections, such as editorial or administrative errors in the referenced ABWR DCD (i.e., misspellings, incorrect references, table headings, etc.). NRC staff reviewed the STD DEP Admin related to the administrative departure included in Section 19.4.5 of the STP Units 3 and 4 COL FSAR. The administrative departure entails minor editorial corrections in the referenced ABWR DCD (e.g., misspellings, etc.) and does not affect the presentation of any probabilistic design discussion. Therefore, this departure is reasonable.

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

Supplemental Information

Subsection 19.4.3.2.1 Structure Fragility

Because of the reclassification of the RWB from Seismic Category 1 to non-seismic in the Departure STD DEP T1 2.15-1, no seismic fragility for this building is needed. The NRC staff's evaluation of this departure is presented above. The staff agreed with the applicant's assertion of no text changes regarding the structure fragility aspects of Departure STD DEP T1.2.15-1 as acceptable.

Subsection 19.4.3.4 Results of the Analysis

NRC staff reviewed the conformance of Section 19.4.3 of the STP Units 3 and 4 COL FSAR to the guidance in RG 1.206, Section C.I.19, "Probabilistic Risk Assessment and Severe Accident Evaluation." The staff's review confirmed that the applicant has addressed the required information related to the "Seismic Margins Analysis." Specifically, the staff concluded that the information pertaining to the STP Units 3 and 4 COL FSAR Tier 2, Revision 2, Section 19.4.3, "Seismic Margins Analysis," is within the scope of the DC and the section adequately incorporates by reference Section 19.4.3 of the ABWR DCD, Revision 4.

Section 19.4.4 Fire Protection Probabilistic Risk Assessment

The applicant reviews the impact of proposed plant departures on the results of the ABWR DCD FIVE analysis. The applicant concludes that the existing ABWR FIVE results bound the STP Units 3 and 4 fire analysis. See Appendix 19M.4 for a discussion of the fire protection PRA.

Section 19.4.5 ABWR Probabilistic Flooding Analysis

This section summarizes the important aspects of the probabilistic flood analysis of the relocated RSW pump house developed in Appendix 19R of Chapter 19 of the STP Units 3 and 4 COL FSAR. NRC staff determined that this section sufficiently summarizes the important aspects of this probabilistic flood analysis developed in Appendix 19R. Based on this finding and on the staff's safety evaluation of Appendix 19R associated with this probabilistic flood analysis, the staff concluded that the supplemental information in Section 19.4.5 associated with the probabilistic flood analysis of the relocated RSW pump house is acceptable.

Section 19.4.5 of COL FSAR Revision 3 summarized the probabilistic flooding analysis for external flooding that was developed under Appendix 19R of the STP Units 3 and 4 COL FSAR Revision 3. However, as a result of the open item identified under Appendix 19R that was associated with this probabilistic flooding analysis, the staff was unable to finalize the conclusions relating to the supplemental information in Section 19.4.5 associated with the probabilistic flooding analysis for external flooding. This issue was tracked as Open Item 19-12 (RAI 19-30) in the SER with open items. The applicant's response to Open Item 19-12 (RAI 19-30) dated July 28, 2010 (ML102110184), states that FSAR Section 19.4.5 will be revised by deleting the discussion related to external flooding. Based on (1) the change in the watertight door status to be normally closed, and (2) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-12 (RAI 19-30) have been resolved. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR. Therefore, the staff found the applicant's modeling of external floods acceptable.

The staff also noted that Departure STD DEP 12.3-3 ("Steam Tunnel Blowout Panel") could impact the results of the PRA flooding analysis. The staff issued RAI 19.01-21 asking the applicant to provide this information. The applicant's response to this RAI dated August 5, 2009 (ML092220163), states that the steam tunnel is designed to handle the consequences of a high-energy pipe break. The steam tunnel is vented to the turbine building. Therefore, any flooding originating in the steam tunnel will end up in the turbine building. The design-basis flood analysis of the turbine building evaluated the consequence for floods originating in the circulating water system (CWS) and the turbine building service water system (TSW). Because the amount of the water caused by the steam tunnel blowout panel is much less than the amount originating from the CWS and TWS floods, the consequence of the flood from the steam tunnel is much smaller. The staff found this approach acceptable. Based on the above discussion, RAI 19.01-21 is resolved.

Section 19.4.6 ABWR Shutdown Risk

As part of the response to RAI 19.01-31 dated February 16, 2011 (ML110490542), the applicant augmented a commitment (COM 19.4-1) to provide specific compensatory actions in FSAR Section 19.4.6. The applicant will develop a hurricane abnormal operating procedure for STP Units 3 and 4 consistent with NUMARC 87–00, Revision 1, "Guidelines and the Technical Bases for NUMARC Initiatives Addressing a Station Blackout at Light Water Reactors," Initiative 2, "Procedures," and Section 2.11, "Hurricane Preparations." The staff found the applicant's commitment acceptable. Refer to SER Section 19L.4 for further discussion on the technical evaluation of this issue.

19.4.5 Post Combined License Activities

The applicant identifies the following commitment as part of the response to RAI 19.01-31:

Commitment (COM 19.4-1, CR 10-15528, Action 2) - Develop a STP 3 and 4 abnormal operating procedure for severe weather that is consistent with NUMARC 87-00, Revision 1, "Guidelines and the Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Initiative 2, "Procedures," and Section 2.11, "Hurricane Preparation," with the following specific requirements as follows:

- Action shall be initiated to place the units in Mode 3 (Hot Shutdown) at least two hours prior to wind speeds in excess of 73 mph (or 96 mph as determined by discussions with the Transmission Distribution Service Provider [TDSP]). The applicability for this requirement is for units in Modes 1 and 2. Units in Modes 3, 4, or 5 will be maintained in Modes 3, 4, or 5.

- One emergency diesel generator (EDG) in each unit is started and loaded onto its safety bus and the bus is disconnected from offsite power at least two hours prior to the arrival onsite of winds in excess of 73 mph.

- If an unstable electrical grid develops or is predicted by the TDSP, the remaining diesel generators are started and loaded on their safety buses and the buses disconnected from offsite power.

- If applicable to the current unit mode, the RCIC will be verified to be available to provide core cooling in the event of a station blackout.

- The portable diesel driven fire pump will be staged in an onsite Seismic Category I structure prior to the arrival onsite of winds in excess of 73 mph.

- If the containment is inerted at the time of the hurricane warning, it will remain inerted during a forced shutdown due to a hurricane, in anticipation of restoring the units to operation after the hurricane has passed.

19.4.6 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 relating to "External Event Analysis and Shutdown Risk Analysis." 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 "External Event Analysis and Shutdown Risk Analysis" that were incorporated by reference have been resolved. In addition, based on the above discussion on "External Event Analysis and Shutdown Risk Analysis," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.4S PRA Maintenance

19.4S.1 Introduction

The applicant describes the STP Units 3 and 4 PRA maintenance and upgrade programs during the COL review, construction, and operational phases.

19.4S.2 Summary of Application

In Section 19.4S, the applicant proposes the following commitments:

> Develop procedures that control the development and maintenance of the as-designed, as-to-be-built, plant-specific PRA during the COL application review phase. This procedure will be used during the construction phase of STP Units 3 and 4. (COM 19.4S-1).
>
> Develop and implement procedures to control the plant walkdown process and identify spatial interactions for the purpose of developing the plant's fire PRA, the internal flooding PRA, and the seismic PRA during the construction phase. (COM 19.4S-2).
>
> Develop and implement procedures similar to those used to control the STP Units 1 and 2 PRA before construction begins (maintenance and update) during

the operations phase to control the incorporation of changes to the as-designed, as-to-be-built plant PRA. (COM 19.4S-3).

Perform an industry peer review of the as-constructed, plant-specific PRA at least 6 months before fuel loading to ensure that the PRA contains the appropriate scope, level of detail, and technical adequacy consistent with the prevailing PRA standards, guidance, and good industry practices. (COM 19.4S-4).

In addition, the applicant states that an existing plant procedure for STP Units 1 and 2 on the PRA Model Maintenance and Update will be used to maintain the plant-specific PRA developed to support operation of STP Units 3 and 4.

19.4S.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in section 19.1.3 of this SER. In particular, this supplemental section of the STP Chapter 19 addressed the PRA quality guidance as described in RG 1.200 and PRA maintenance and upgrade guidance described in RG 1.206, Section C.I.19.7.

Also, 10 CFR 50.71(h)(1) states that no later than the scheduled date for initial loading of fuel, each holder of a COL shall develop a level 1 and level 2 PRA. In addition, 10 CFR 50.71(h)(2) states that each holder of a COL shall maintain and upgrade the PRA required by 10 CFR 50.71(h)(1). The upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade. The PRA must be upgraded every four years until the permanent cessation of operations.

19.4S.4 Technical Evaluation

The applicant commits to the NRC to develop procedures that control the development and maintenance of the as-designed, as-to-be-built, plant-specific PRA during the COL application review phase (COM 19.4S-1). This procedure will be used during the construction phase of STP Units 3 and 4.

The applicant commits to develop and implement procedures that control the plant walkdown process and identify spatial interactions for the purpose of developing the plant fire PRA, the internal flooding PRA, and the seismic PRA during the construction phase (COM 19.4S-2). The applicant commits to develop and implement procedures similar to those used to control the STP Units 1 and 2 PRA (1) before construction begins (maintenance and update), and (2) during the operations phase to control the incorporation of changes to the as-designed, as-to-be-built plant PRA (COM 19.4S-3). The staff issued an RAI 19.01-26 requesting the applicant to clarify whether the procedures the applicant has developed will be used in the operational phase.

The applicant's response to RAI 19.01-26 dated August 5, 2009 (ML092220163), states that STP Units 3 and 4 will develop and implement procedures, before the start of construction, similar to those used to control the STP Units 1 and 2 PRA maintenance and update during the operations phase to control the incorporation of changes to the as-designed, as-to-be-built plant PRA. The staff found this response acceptable, and this RAI is resolved. Verification that the proposed revision is incorporated into Revision 4 of the FSAR was being tracked as Confirmatory Item 19-5 in the SER with open items. The staff confirmed that the proposed revision is incorporated into Chapter 19 of FSAR Revision 4. RAI 19-7 is therefore resolved

The applicant commits to perform an industry peer review of the as-constructed, plant-specific PRA at least 6 months before fuel loading to ensure that the PRA contains the appropriate scope, level of detail, and technical adequacy consistent with the prevailing PRA standards, guidance, and good industry practices (COM 19.4S-4).

The staff reviewed Section 19.4S of the STP Units 3 and 4 COL and checked the referenced DCD. This new section satisfies the PRA maintenance and upgrade guidance described in RG 1.206, Section C.I.19.7.

19.4S.5 Post Combined License Activities

The applicant identifies the following commitments:

- Commitment (COM 19.4S-1) – Develop procedures that control the development and maintenance of the as-designed, as-to-be-built, plant-specific PRA during the COL application review phase.

- Commitment (COM 19.4S-2) – Develop and implement procedures to control the plant walkdown process to identify spatial interactions for the purpose of developing the plant fire PRA, the internal flooding PRA, and the seismic PRA during the construction phase.

- Commitment (COM 19.4S-3) – Develop and implement procedures, before construction starts to control the incorporation of changes to the as-designed, as-to-be-built plant PRA.

- Commitment (COM 19.4S-4) – Perform an industry peer review of the as-constructed plant-specific PRA at least 6 months before fuel loading to ensure that the PRA contains the appropriate scope, level of detail, and technical adequacy consistent with the prevailing PRA standards, guidance, and good industry practices.

19.4S.6 Conclusion

NRC staff reviewed the application and checked the reference DCD. This section is a supplement to the original DCD. No outstanding information is expected to be addressed in the COL FSAR related to this section. Based on the above discussion on “PRA Maintenance,” the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.5 Source Term Sensitivity Studies (Related to RG 1.206, Part I, C.I.19, Appendix A, Subsection 19.1.4.1.1, “Description of the Level 1 PRA for Operation at Power.”)

Section 19.5 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.5, “Source Term Sensitivity Studies,” of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding information outside of the DCD 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 "Source Term Sensitivity Studies" have been resolved.

19.6 Measurement Against Goals (Related to RG 1.206, Part I, C.I.19, Appendix A, Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation"; Section 19.1.3, "Special Design/Operational Features"; and Subsection 19.1.4.1.1, "Description of the Level 1 PRA for Operation at Power.")

19.6.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.6 of the U. S. ABWR DCD due to a minor reference change of the STP Units 3 and 4.

19.6.2 Summary of Application

Section 19.6 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.6 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, the applicant provides the following in FSAR Section 19.6:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP ADMIN

This departure corrects a cross-reference between sections of the ABWR DCD and the SSAR.

19.6.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in section 19.1.3 of this SER.

19.6.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 19.6 of the certified ABWR DCD. The staff reviewed Section 19.6 of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Measurement Against Goals."

The staff reviewed the information in the COL FSAR:

[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 2 Departure Not Requiring Prior NRC Approval

- STD DEP Admin

The applicant defines administrative departures as minor corrections, such as editorial or administrative errors in the referenced ABWR DCD (e.g., misspellings, incorrect references, table headings, etc.). The applicant points to Subsection 19.D.5.2 of the DCD, "Accident Classes," (2) Class II to note that there was substantial time available (about 24 hours) to repair any heat removal systems that initially fail.

The applicant's evaluation of this departure described above, in accordance with Item B.5 of Section VIII, determined that this departure does not require prior NRC approval. The staff reviewed this departure, and could not determine whether it is reasonable for this departure not to require prior NRC approval. Therefore, the staff issued an RAI 19.01-27, Question 1, asking the applicant to clarify that Subsection 19.D.5.2 refers to the ABWR SSAR. The applicant confirmed that the information is in the ABWR SSAR. Therefore, this RAI is resolved. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval.

19.6.5 Post Combined License Activities

There are no COL license information items in this section.

19.6.6 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, 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 "Measurement Against Goals" that were incorporated by reference have been resolved.

19.7 PRA as a Design Tool (Related to RG 1.206, Part I, C.I.19, Appendix A, Subsections 19.1.1.1, "Uses and Applications of the PRA"; 19.1.2.1, "PRA Scope"; 19.1.7.1, "PRA Input to Design Programs and Processes"; Section 19.1.3, "Special Design/Operational Features"; and Section 19.2, "Severe Accident Evaluation.")

19.7.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.7 of the U. S. ABWR DCD due to the departures of the STP Units 3 and 4 design from those described in the ABWR DCD.

19.7.2 Summary of Application

Section 19.7 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.7 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, the applicant provides the following in FSAR Section 19.7:

Tier 1 Departure

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

This departure eliminates obsolete data communication technology and unnecessary inadvertent actuation prevention logic and equipment. The departure also changes the implementation, architecture, testing, and surveillance descriptions for the SSLC.

Tier 2 Departure Requiring Prior NRC Approval

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

This departure addresses using two MVES (13.4 kV and 4.6 kV) instead of the one 6.9 kV MVES described in the ABWR DCD. This departure affects Section 19.7.3, “PRA Studies During the Certification Effort,” by changing the output voltage design of the combustion turbine generator (CTG) and the electrical loads supported by this generator.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 19.7-1 Control Rod Drive Improvements

This departure addresses the fine motion control rod drive (FMCRD) brake design testing. The ABWR DCD states that the FMCRD brake design had to be fully testable on an annual basis to meet the goals for rod ejection frequency. The annual test frequency assumes that the plant is operating under an annual cycle and the inspection is conducted during an outage. For plants operating in an 18-month cycle, testing the brakes during power operation is not practical. Section 19.7.2, “Early PRA Studies,” clarifies the consistency relating to outages on the 18-month cycle basis for the plant. The applicant states that the FMCRD brake design has to be fully testable on a refueling cycle basis, and the words “refueling cycle” replace the words “an annual.”

- STP DEP 19R-1 Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

This departure addresses the internal flooding of the control building due to the elimination of vacuum breaker valves on the supply and return piping connecting to the RBCW heat exchangers. Elimination of the vacuum breaker valves is due to the RSW system design changes that include the use of horizontal type pumps instead of vertical wet-pit type pumps and piping configuration changes between the UHS basin and the control building. This departure affects Section 19.7.3, “PRA Studies During the Certification Effort,” by eliminating the need for considering the anti-siphon capability and pipe length limit in the RSW design.

Supplemental Information

Subsection 19.7.2 Early PRA Studies

The text changes are the results of Departures STD DEP T1 3.4-1 and STD DEP 19.7-1.

Subsection 19.7.3 PRA Studies During the Certification Effort

The text changes reflected the Departures STD DEP 8.3-1 and STD DEP 19R-1.

19.7.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19.7.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 19.7 of the certified ABWR DCD. The staff reviewed Section 19.7 of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "PRA as a Design Tool."

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 their technical impact may be evaluated in the 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 Tier 1 departure.

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

The evaluation is described in Section 19.1.4 of this SER.

Tier 2 Departure Requiring Prior NRC Approval

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

The evaluation is described in Section 19.3.4 of this SER.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 19.7-1 Control Rod Drive Improvements

The technical evaluation of this departure is documented in Section 4.6 of this SER. The change in testing frequency in Section 19.7.2, Item 4, of the referenced ABWR DCD is proposed to reflect that the plant's refueling outage will be every 18 months, during which time the FMCRD brakes can be tested. This departure does not affect the brake design or function. The testing is to assure that the brake performance to prevent rod ejection is not affected, as considered in the ABWR PRA studies. The change in the brake testing frequency description does not impact the brake design or function and therefore, the likelihood or consequence of a severe accident is not affected. Therefore, the staff found the supplemental information 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.

- STP DEP 19R-1 Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

The technical evaluation of this departure is documented in Section 9.2.15 of this SER. The text deletions in Section 19.7.3, Item 4, paragraph 5 (third bullet) appropriately reflect the RSW design changes under this departure. These changes include the use of horizontal-type pumps instead of vertical, wet-pit type pumps and piping configuration changes between the UHS basin and the control building. The impact of these RSW design changes on plant risk is evaluated in Appendix 19R of Chapter 19 of the STP Units 3 and 4 COL FSAR. The staff found these changes acceptable.

Supplemental Information

Subsection 19.7.2 Early PRA Studies

The text changes in "Instrumentation Studies" are the results of Departure STD DEP T1 3.4-1 and text changes from "annual basis" of Control Rod Drive brake inspection to "refueling cycle" basis was evaluated in this Section as the result of STD DEP 19.7-1. The changes are editorial in nature. Therefore, the supplemental information is this section is acceptable.

Subsection 19.7.3 PRA Studies During the Certification Effort

The text changes in "Combustion Turbine Generator" reflect that the medium voltage system is changed from 6.9KV stated in the ABWR DCD to 4.16KV (Departure STD DEP 8.3-1). The text deletions related to the discussion of the RSW in this section of the FSAR are the result of Departure STD DEP 19R-1. The changes are editorial in nature. Therefore, the supplemental information is acceptable.

19.7.5 Post Combined License Activities

The applicant identifies commitment (COM 19.9-14) to address COL License Information Item 19.15 as discussed in SER Section 19.9.4.

19.7.6 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 NRC staff's review confirmed that the applicant has addressed the required information relating to "PRA as a Design Tool." 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 "PRA as a Design Tool" that were incorporated by reference have been resolved.

In addition, based on the above discussion on "PRA as a Design Tool," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.8 Important Features Identified by the ABWR PRA (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.3, "Special Design/Operational Features"; 19.1.4.2, "Level 2, Internal Events PRA for Operations at Power"; 19.1.6.2, "Results from the Low-Power and Shutdown Operations PRA"; 19.1.7, "PRA-Related Input to Other Programs and Processes"; 19.1.8, "Conclusions and Findings"; and 19.2, "Severe Accident Evaluation.")

19.8.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.8 of the U. S. ABWR DCD due to the departures of the STP Units 3 and 4 design from the design described in the ABWR DCD.

19.8.2 Summary of Application

Section 19.8 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.8 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in COL FSAR Section 19.8, the applicant provides the following:

Tier 1 Departures

- STP DEP T1 5.0-1 Site Parameters (Table 19.8-5)

This departure addresses the external flooding analysis in Subsection 19.8.5.3.

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

This departure eliminates obsolete data communication technology and unnecessary inadvertent actuation prevention logic and equipment. The departure also changes the implementation, architecture, testing, and surveillance descriptions for the SSLC.

Tier 2 Departure Not Requiring Prior NRC Approval

- STP DEP 19R-1 Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

This departure addresses internal flooding of the control building due to the elimination of vacuum breaker valves on the supply and return piping connected to the RBCW heat exchangers. Elimination of the vacuum breaker valves is due to the RSW system design changes that include the use of horizontal-type pumps instead of vertical wet-pit type pumps, and piping configuration changes between the UHS basin and the control building.

Supplemental Information

Subsection 19.8.5.1 Summary of Analysis Results and Subsection 19.8.5.3 Features Selected

Subsections 19.8.5.1 and 19.8.5.3 of the STP Units 3 and 4 COL FSAR discuss important features identified in the probabilistic flooding analysis of the relocated RSW pump house.

Site-specific supplemental information in Subsection 19.8.5.3 also discusses important features identified in the probabilistic flooding analysis of external flooding, which addresses Departure STP DEP T1 5.0-1 ("Site Parameters").

19.8.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19.8.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 19.8 of the certified ABWR DCD. The staff reviewed Section 19.8 of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Important Features Identified by the ABWR PRA."

The staff reviewed the information in the COL FSAR:

Tier 1 Departures

- STP DEP T1 5.0-1 Site Parameters (Table 19.8-5)

The impact of the Tier 1 departure on the external flooding analysis is addressed in Subsection 19.8.5.3, Table 19.8-5. The applicant states that all external entrances to safety-related buildings located below the maximum flood level have watertight doors or barriers. These measures ensure that no water enters safety-related buildings, thereby allowing a safe shutdown at the plant. The evaluation of this departure on the PRA results is addressed in the Supplemental Information below.

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

The applicant renames the essential multiplexing system to the essential communication function in Table 19.8-1. This change has no impact on the important features identified in the ABWR. The evaluation of this departure is described in Section 19.1.4 of this SER.

Tier 2 Departure Not Requiring Prior NRC Approval

- STP DEP 19R-1 Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

NRC staff reviewed Departure STP DEP 19R-1 included under Section 19.8.5 of the STP Units 3 and 4 COL FSAR. The text that was deleted in Subsection 19.8.5.3 related to the "Anti-siphon Capability"; the "RSW System"; and the "Ultimate Heat Sink." The deletion appropriately reflects the RSW design changes under Departure STP DEP 19R-1. These changes include the use of horizontal-type pumps instead of vertical wet-pit type pumps and piping configuration changes between the UHS basin and control building. The technical impact of these RSW design changes on plant risk is evaluated in Appendix 19R of Chapter 19 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.

Supplemental Information

Subsection 19.8.5.1 Summary of Analysis Results and Subsection 19.8.5.3 Features Selected

NRC staff reviewed the supplemental information related to important features identified in the probabilistic flooding analysis of the relocated RSW pump house, which is included under Section 19.8.5 of the STP Units 3 and 4 COL FSAR.

Section 19.8.5 summarizes the important features identified in the probabilistic flooding analysis of the relocated RSW pump house developed in Appendix 19R of Chapter 19 of the STP Units 3 and 4 COL FSAR. The staff determined that Section 19.8.5 sufficiently summarizes the important features identified in this probabilistic flooding analysis developed in Appendix 19R. Based on this finding and the staff's safety evaluation of Appendix 19R associated with this probabilistic flooding analysis, the staff concluded that the supplemental information in Section 19.8.5 associated with the important features identified in the probabilistic flooding analysis of the relocated RSW pump house is acceptable.

NRC staff reviewed the supplemental information related to important features identified in the probabilistic flooding analysis for external flooding, which is included in Section 19.8.5 of the STP Units 3 and 4 COL FSAR. In response to Open Item 19-12 (see RAI 19-30 in Appendix 19R) dated July 28, 2010 (ML1021101840), the applicant states that FSAR Section 2.4 and Subsection 19.8.5.3 will be revised to state that the flooding analysis assumes that all watertight doors are closed and dogged to prevent the flooding from propagating to another area or from the outside to the inside. The watertight doors are alarmed to alert security personnel that a watertight door is open. However, with the exception of the watertight doors in the RSW pump house, the watertight doors will not alarm to indicate that a door is not dogged. To guard against doors being left undogged, operators should check the doors at every shift to assure that they are closed and dogged.

However, as a result of the open item identified under Appendix 19R that was associated with this probabilistic flooding analysis, the staff was unable to finalize the conclusions relating to the supplemental information in Section 19.8.5 associated with the probabilistic flooding analysis for external flooding. This issue was tracked as Open Item 19-12 (RAI 19-30) in the SER with open items. Based on (1) the change in watertight door status to be normally closed, and (2) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-12 (RAI 19-30) have been resolved. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR. Therefore, the staff found the applicant's modeling of external floods acceptable.

19.8.5 Post Combined License Activities

The applicant identifies commitment (COM 19.9-17) to address COL License Information Item 19.8 as discussed in SER Section 19.9.4.

19.8.6 Conclusion

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Important Features Identified by the ABWR PRA." 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 “Important Features Identified by the ABWR PRA” that were incorporated by reference have been resolved.

In addition, based on the above discussion on “Important Features Identified by the ABWR PRA,” the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.9 COL License Information (Related to RG 1.206, Part I, C.I.19, Appendix A, Sections 19.3.1, “Resolution of Open Items”; 19.3.2, “Resolution of Confirmatory Items”; and 19.3.3, “Resolution of COL Items.”)

19.9.1 Introduction

This section provides responses from the applicant to complete the COL license information items identified in the DCD.

19.9.2 Summary of Application

Section 19.9 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.9 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Section 19.9, the applicant provides the following:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure addresses the pump and turbine monoblock design (the pump and turbine are contained in the same casing), which simplifies the design and removes multiple components.

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

This departure addresses the elimination of obsolete data communication technology and unnecessary inadvertent actuation of prevention logic and equipment. There is a clarification of digital controls nomenclature and systems and a change in implementation architecture and SSLC testing and surveillance.

- STP DEP T1 5.0-1 Site Parameters

This departure addresses the site design-basis flood level, the maximum design precipitation rate for rainfall, and the humidity (wet-bulb temperature). Also, the shear wave velocity at the STP Units 3 and 4 site will not be bounded by the ABWR DCD.

Tier 2 Departures Not Requiring Prior NRC Approval

- STP DEP 9.2-5 Reactor Service Water (RSW) System

This departure addresses the increased RSW flow rate required for the increased heat load in the STP Units 3 and 4 design.

- STD DEP 10.4-5 Condensate and Feedwater System

This departure increases the number of reactor feed pumps from three to four and adds four condensate booster pumps to the system.

- STP DEP 19R-1 — Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

This departure addresses the internal flooding of the control building as a result of the elimination of RSW vacuum breaker valves on the supply and return piping that connects to the RBCW heat exchangers.

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

- COL License Information Item 19.1 — Post Accident Recovery Procedure for Unisolated CUW Line Break

This COL license information item specifies that the applicant develop and implement (before fuel loading) an operating procedure for post accident recovery from a reactor water cleanup system (CUW) line break. (COM 19.9-1).

- COL License Information Item 19.2 — Confirmation of CUW Operation Beyond Design Basis

This COL license information item specifies that the applicant evaluate the CUW operation in the heat removal mode, update the PRA, and develop and implement the emergency operating procedure for operating the CUW in the heat exchanger bypass mode (before fuel loading). (COM 19.9-2).

- COL License Information Item 19.3 — Event Specific Procedures for Severe External Flooding

This COL license information item specifies that the applicant provide site-specific supplemental information in Section 19.9.3 of the STP Units 3 and 4 COL FSAR, for developing and implementing an operating procedure for external flooding before fuel loading. There are also guidelines for this procedure. (COM 19.9-3).

- COL License Information Item 19.4 — Confirmation of Seismic Capacities Beyond the Plant Design Basis

This COL license information item specifies that the applicant complete the seismic capacity analysis before fuel loading. (COM 19.9-4).

- COL License Information Item 19.5 — Plant Walkdowns

This COL license information item specifies that the applicant develop before fuel loading procedures for plant walkdowns to identify seismic, fire, and internal flooding vulnerabilities. (COM 19.9-5).

- COL License Information Item 19.6 — Confirmation of Loss of AC Power Event

This COL license information item specifies that the applicant provide an assessment that addresses site-specific parameters, such as specific causes of the loss of power and their impact on a timely recovery of AC power.

- COL License Information Item 19.7 — Procedures and Training for Use of AC Independent Water Addition

This COL license information item specifies that the applicant develop and implement operating procedures and training for AC-Independent Water Addition (ACIWA). These procedures will identify system valve actuations that provide ACIWA via the RHR system, as a water source to the RPV or to the containment. (COM 19.9-6).

- COL License Information Item 19.8 Actions to Avoid Common Cause Failures in the Essential Communications Function (ECF) and Other Common Cause Failures

This COL license information item specifies that the applicant develop and implement test, maintenance, surveillance, and administrative procedures before fuel loading to ensure that credible common mode failures cannot occur. (COM 19.9-7).

- COL License Information Item 19.9 Actions to Mitigate Station Blackout Events

This COL license information item specifies that the applicant develop analyses and procedures (before fuel loading) to confirm the assumptions modeled in the PRA. (COM 19.9-8).

- COL License Information Item 19.10 Actions to Reduce Risk of Internal Flooding

This COL license information item specifies that the applicant provide site-specific supplemental information for developing and implementing (before fuel loading) training, design, a site-specific PRA-based analysis, and procedures to reduce the risk of internal flooding. (COM 19.9-9).

- COL License Information Item 19.11 Actions to Avoid Loss of Decay Heat Removal and Minimize Shutdown Risk

This COL license information item specifies that the applicant develop and implement (before fuel loading) operating procedures to avoid the loss of decay heat removal during a shutdown condition. (COM 19.9-10).

- COL License Information Item 19.12 Procedures for Operation of RCIC from Outside the Control Room

This COL license information item specifies that the applicant develop procedures and conduct training for the RCIC operation. (COM 19.9-11)

- COL License Information Item 19.13 ECCS Test and Surveillance Intervals

This COL license information item specifies that the applicant provide standard supplemental information for developing and implementing (before fuel loading) a plan and procedures to identify departures from the test and surveillance intervals assumed in Tables 19D.6-1 through 19D.6-12. (COM 19.9-12).

- COL License Information Item 19.14 Accident Management

This COL license information item specifies that the applicant include operator actions in the operating and training procedures to be developed and implemented before fuel loading. (COM 19.9-13).

- COL License Information Item 19.15 Manual Operation of MOVs

This COL license information item specifies that the applicant develop and implement before fuel loading a procedure for operating MOVs manually. (COM 19.9-14).

- COL License Information Item 19.16 High Pressure Core Flooder Discharge Valve

This COL license information item specifies that the applicant develop and implement a procedure for verifying that the high pressure core flooder (HPCF) discharge valve is in the locked-open position before fuel loading. (COM 19.9-15).

- COL License Information Item 19.17 Capability of Containment Isolation Valves

This COL license information item specifies that the applicant demonstrate before fuel loading that the containment isolation valves will not exceed ASME Section III Service Level C limits and the ultimate pressure capability of the valves will be greater than 1.03 MPa. (COM 19.9-16).

- COL License Information Item 19.18 Procedure to Ensure Sample Lines and Drywell Purge Lines Remain Closed During Operation

This COL license information item specifies that the applicant develop operating procedures and administrative controls to ensure that sample lines and drywell purge lines will remain-closed during operation. (COM 19.9-17).

- COL License Information Item 19.19 Procedures for Combustion Turbine Generator to Supply Power to Condensate and Condensate Booster Pumps

This COL license information item specifies that the applicant develop and implement before fuel loading operating procedures for manually transferring the CTG power to the condensate, condensate booster pumps, and the support systems. (COM 19.9-18).

- COL License Information Item 19.19a Actions to Assure Reliability of the Supporting RCW and Service Water Systems

This COL license information item specifies that the applicant develop and implement before fuel loading operating procedures for swapping RCW and RSW operating pumps and heat exchangers at least monthly. (COM 19.9-19).

- COL License Information Item 19.19b Housing of ACIWA Equipment

This COL license information item specifies that the applicant demonstrate (before fuel loading) the capability of the building that houses the ACIWA equipment to withstand site-specific seismic events, flooding, and other site-specific external events that will be confirmed and included in the plant-specific PRA. (COM 19.9-20).

- COL License Information Item 19.19c Procedures to Assure SRV Operability During Station Blackout

This COL license information item specifies that the applicant develop and implement (before fuel loading) operating procedures for aligning stored nitrogen bottles for the SRVs. (COM 19.9-21).

- COL License Information Item 19.19d Procedures for Ensuring Integrity of Freeze Seals

This COL license information item specifies that the applicant develop and implement (before fuel loading) procedures for using and administratively controlling freeze seals. (COM 19.9-22).

- COL License Information Item 19.19e Procedures for Controlling Combustibles During Shutdown

This COL license information item specifies that the applicant develop and implement (before fuel loading) administrative procedures for controlling combustibles and ignition sources. (COM 19.9-23).

- COL License Information Item 19.19f Outage Planning and Control

This COL license information item specifies that the applicant develop and implement (before fuel loading) an outage planning and control program that is consistent with NUMARC 91-06 criteria. (COM 19.9-24).

- COL License Information Item 19.19g Reactor Service Water Systems Definition

This COL license information item addresses the overall results of the STP RSW and considers the effect of Departure STP DEP 9.2-5.

- COL License Information Item 19.19h Capability of Vacuum Breaker

This COL license information item specifies that the applicant demonstrate (before fuel loading) the capability of the vacuum breaker seating material to withstand the temperature profiles associated with the equipment survivability requirements specified in Subsection 19E.2.1.2.3. (COM 19.9-25).

- COL License Information Item 19.19i Capability of the Containment Atmospheric Monitoring System

This COL license information item addresses the requirement that the containment atmospheric monitoring (CAM) system can be exposed to containment pressures consistent with the loading associated with the equipment survivability requirements specified in Subsection 19E.2.1.2.3 before fuel loading. (COM 19.9-26).

- COL License Information Item 19.19j Plant Specific Safety-Related Issues and Vendors Operating Guidance

This COL license information item specifies that the applicant develop (before fuel loading) plant operating procedures for maintaining important safety functions during shutdown operations. (COM 19.9-27).

- COL License Information Item 19.30 PRA Update

This COL license information item addresses the overall results. The applicant indicated that the PRA evaluation is bounded by the conclusions of the standard ABWR DCD, Subsection 19.3.1.5, “Results in Perspective.”

19.9.3 Regulatory Basis

The relevant requirements for the Commission’s regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

In addition, RG 1.206, Part III, Section C.III.4.3 provides guidance and requests that the applicant describe the implementation schedules and plans for the resolution of the COL licensing information.

19.9.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 19.9 of the certified ABWR DCD. The staff reviewed Section 19.9 of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL 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 "COL License Information."

The staff reviewed the information in the COL FSAR:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump Design

The pump and turbine are a monoblock design (the pump and turbine are contained in the same casing), which simplifies the design and removes multiple components See evaluation in Section 19.3.4 of this SER.

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

Departure STD DEP T1 3.4-1 can be characterized as five primary changes, two of which impact Section 19.9.8:

1. Elimination of references to the essential multiplexer system (EMS) and the non-essential multiplexer system (NEMS) originally envisioned in the ABWR architecture; these references are replaced with separate and independent system level data communication capabilities.

2. Clarification of digital controls nomenclature and systems.

The staff determined that these specific text changes are appropriate and address Departure STD DEP T1 3.4-1. See evaluation in Section 19.1.4 of this SER.

- STP DEP T1 5.0-1 Site Parameters

The site design-basis flood level, the maximum design precipitation rate for rainfall, the humidity (represented by wet-bulb temperature), and the shear wave velocity at the STP site are not bounded by the descriptions in the ABWR DCD.

Departure STP DEP T1 5.0-1 also impacts the external flooding analysis developed in Appendix 19R, of Chapter 19 of the STP Units 3 and 4 COL FSAR, which is used in Section 19.9.3 to develop guidelines for event-specific procedures for external flooding. NRC staff determined that the departures under Section 19.9.3 appropriately reflect Departure STP DEP T1 5.0-1, as well as the departures related to the external flooding analysis under Appendix 19R. However, as a result of the open item identified in Appendix 19R that was

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.

associated with the external flooding analysis, the staff was unable to finalize the conclusions for these departures in Section 19.9.3. This issue was tracked as Open Item 19-12 (RAI 19-30) in the SER with open items. Based on (1) the change in watertight door status to be normally closed, and (2) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-12 (RAI 19-30) have been resolved. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR. Therefore, the staff found the applicant's modeling of external floods acceptable.

Tier 2 Departures Not Requiring Prior NRC Approval

- STP DEP 9.2-5 — Reactor Service Water (RSW) System

The STP COL states that the RSW and UHS have been modified to meet the increased heat removal requirements of the RCW system for STP Units 3 and 4. The potential impact is included in the delta-PRA analysis. The impact of these RSW design changes on plant risk is evaluated in Appendix 19R of Chapter 19 of the STP Units 3 and 4 COL FSAR.

- STD DEP 10.4-5 — Condensate and Feedwater System

The departure adds an additional reactor feed pump, two heater drain pumps, and four condensate booster pumps to this system. See evaluation in Section 19.1.4.

- STP DEP 19R-1 — Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

This departure eliminates vacuum breaker valves on the supply and return piping that connects to the RCW heat exchangers evaluated in the ABWR SSAR that were added to the STP COL application. The impact of these RSW design changes on plant risk is evaluated in Appendix 19R of Chapter 19 of the STP Units 3 and 4 COL FSAR.

COL License Information Items

- COL License Information Item 19.1 — Post Accident Recovery Procedure for Unisolated CUW Line Break

In Section 19.9.1, the applicant commits (COM 19.9-1) to develop and implement (before fuel loading) an operating procedure for the post accident recovery from a CUW line break. This commitment contains the provisions for procedure development that was described in the COL license information item in the DCD.

NRC staff reviewed the proposed commitment (including procedure development provisions) in the FSAR, and also examined the COL license information in the DCD, as well as the evaluation of this COL license information item in the ABWR DCD FSER. The staff found that the proposed commitment contains sufficient information for procedure development and is acceptable.

- COL License Information Item 19.2 — Confirmation of CUW Operation Beyond Design Basis

In Section 19.9.2, the applicant commits (COM 19.9.2) to complete an evaluation of the CUW operation in the heat removal mode, update the PRA before fuel loading, and develop and implement the emergency operating procedure for operating the CUW in the heat exchanger bypass mode before fuel loading.

NRC staff issued RAI 19-15, which asked how the applicant will complete and track the evaluation of the CUW operation in the heat removal mode and the PRA update.

The applicant's response to RAI 19-15 dated July 13, 2009 (ML092740559) notes that Section 19.9.2 of the STP COL application, Tier 2 will be revised to state that an evaluation of the CUW operation in the heat removal mode will be completed before fuel loading (COM 19.9-28). The applicant also states that this evaluation will confirm that areas listed in STP FSAR Section 19.9.2 will remain functional while operating outside of their design-basis temperature values. The staff found that this response to RAI 19-15 is sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. Verification that the proposed revision is incorporated into Revision 4 of the FSAR was tracked as Confirmatory Item 19-7 in the SER with open items. The staff confirmed that the proposed change is incorporated into Chapter 19 of FSAR Revision 4. Therefore, RAI 19-15 is resolved.

- COL License Information Item 19.3 Event Specific Procedures for Severe External Flooding

In Section 19.9.3, the applicant commits to develop and implement (before fuel loading) an operating procedure for external flooding. (COM 19.9-3).

NRC staff determined that the supplemental information in Section 19.9.3 is also consistent with the external flooding analysis developed under Appendix 19R. However, as a result of the open item identified under Appendix 19R that was associated with the external flooding analysis, the staff was unable to finalize the conclusions for the supplemental information in Section 19.9.3. This issue was tracked as Open Item 19-12 (RAI 19-30) in the SER with open items. In response to RAI 19-30 dated July 28, 2010 (ML102110184), the applicant states that FSAR Section 19.9.3 will be revised to state that the event-specific procedures for severe external flooding assume that all watertight doors are closed and dogged to prevent floods from propagating to another area or from the outside to the inside. The watertight doors are alarmed to alert security personnel that a watertight door is open. However, with the exception of the watertight doors in the RSW pump house, the watertight doors will not alarm to indicate that a door is not dogged. To guard against doors being left undogged, operators should check the doors at every shift to assure that they are closed and dogged. Procedures and training will be developed to ensure that observation of the main cooling reservoir is conducted so that main control room personnel will be alerted by indications of a main cooling reservoir breach. These procedures will direct that all watertight doors be verified closed immediately upon the receipt of such notification.

Based on (1) the change in watertight door status to be normally closed, and (2) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-12 (RAI 19-30) have been resolved. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR. Therefore, the staff found the applicant's modeling of external floods acceptable.

- COL License Information Item 19.4 Confirmation of Seismic Capacities Beyond the Plant Design Basis

In Section 19.9.4, the applicant commits (COM 19.9-4) to complete the seismic capacity analysis before fuel loading. COL License Information Item 19.9.4 in the ABWR DCD, Revision 4, calls for the implementation of actions specified in Section 19H.5.1, including the need for an evaluation of the site-specific plant level high confidence low probability of failure (HCLPF) capacity of the generic SSCs, which are not part of the standard ABWR SSCs and whose fragilities were assumed based on typical component designs. The list of generic components in Section 19H.4.3 includes the plant-specific, safety related SSCs (e.g., piping and service water pump house). The applicant's statement in Section 19.9.4 of STP Units 3 and 4 COL FSAR that the seismic capacity analysis will be completed before fuel loading and the PRA will be updated in accordance with 10 CFR 50.71(h)(1), lacks the necessary details to adequately address COL License Information Item 19.4. NRC staff issued RAI 19-27 requesting the applicant to discuss in detail and elaborate how items listed in Section 19H.5.1 will be implemented, especially "Step 3 - Assessment of As-Built SMA SSC HCLPF Values" of the ABWR DCD. The applicant's response to RAI 19-27 dated August 26, 2009 (ML092430135), identifies the following revisions and additions to the FSAR COL application, Section 19.9.4, which will be revised as follows in a future update:

> 19.9.4 Confirmation of Seismic Capacities beyond the Plant Design Basis
>
> The following standard supplement addresses COL License Information Item 19.9-4). The seismic capacity analysis will be completed prior to fuel loading and the PRA will be updated in accordance with 10 CFR 50.71(h)(1), (COM 19.9-4). The following actions will be taken (COM 19.9-4):
>
> 1. The High-Confidence Low Probability of Failure (HCLPF) values for the important plant specific/as-built components corresponding to the generic components defined in Subsection 19H.4.3 shall be determined. The values will be compared to the assumed HCLPF values given in Tables 19H-1 or 19I-1. This will be completed prior to fuel load.
>
> 2. HCLPF values will be established for site-specific SSCs (UHS/pump house structure and cooling tower) that are not included in the analyses described in Appendix 19H and whose failure may affect the plant response to seismic events.
>
> 3. The investigation for the potential for seismic induced soil failure at 1.67 times the site specific ground motion response spectra (GMRS) will be completed prior to fuel load.
>
> 4. The remainder of the actions specified in Appendix 19H.5 will be completed prior to fuel load.

The staff noted that, in the response to RAI 19-27, the applicant identifies the UHS/pump house structure and cooling tower as items not explicitly included among the generic SSCs in Appendix 19H but needing to be analyzed as part of the plant-specific Category I structures. ABWR DCD, Revision 4, Section 19.9.26, "Reactor Service Water Systems Definition," directs the COL applicant to review RSW and UHS design configurations and performance capabilities against those assumed and modeled in the DCD and SSAR. The RSW system consists of piping, tunnel structures, and connections to the pump house and control building. Therefore, the applicant's response was considered incomplete and needed to be augmented. The staff issued supplemental RAI 19-29 asking the applicant to include and describe the complete set of SSCs that make up the UHS/RSW system under Action Item 2 above. In a letter dated January 14, 2010 (ML100190245), the applicant responded to this request by explicitly including the RSW system under Item 2 of the future revision to the COL FSAR Section 19.9.4. This supplemental RAI is therefore resolved.

The staff found the applicant's responses to RAI 19-27 and RAI 19-29 adequate and acceptable. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR. Therefore, these RAIs are closed.

ABWR DCD, Section 19H.5.1 requires the soil liquefaction evaluation and slope stability analysis be performed for 1.67 times the site-specific safe-shutdown earthquake (SSE). In RAI 19-25, the staff requested the applicant to confirm that such an evaluation will be performed or provide the basis for not performing the evaluation. The staff found that the applicant's response to RAI 19-25 dated August 26, 2009 (ML092430135), confirms that an analysis for a potential liquefaction induced failure will be performed for 1.67 times the site-specific GMRS before fuel loading. Furthermore, the applicant states that there are no safety-related slopes at STP Units 3 and 4. The staff determined that this response is satisfactory and is in accordance with the ABWR DCD FSER (NUREG–1503) and COL Commitment 19.9-4. RAI 19-25 is resolved and closed.

In accordance with ABWR DCD COL License Information Item 19.9.4, the applicant is directed to evaluate the HCLPF capacities of standard plant and site-specific SSCs for updating the PRA. In RAI 19-31, the staff requested the applicant to confirm that this COL license information item includes an update of the system model (seismic accident sequences) developed in the DCD to incorporate capacity reductions due to site-specific effects (soil liquefaction, slope failure, etc.) and site-specific SSCs (the UHS and the RSW including the pump house, cooling tower, and water reservoir), and to determine whether site-specific soil failures control the seismic HCLPF capacities of SSCs associated with the seismic accident sequences. Based on the results of the update, the applicant was also requested to demonstrate the sequence-level and plant-level seismic HCLPF capacity. The staff needed this information to ensure that the applicant's PRA-based SMA complies with pertinent requirements of 10 CFR 52.79(a)(46) and 10 CFR 52.79(d)(1). This RAI was tracked as Open Item 19-17.

The applicant's response dated May 04, 2010 (ML101260119) confirmed that the system model (seismic accident sequences) developed in the DCD will be updated to incorporate seismic capacity reductions due to site-specific effects (soil liquefaction) and site-specific SSCs (the UHS including the RSW pump house, cooling tower, RSW piping tunnel, and diesel generator oil storage vault). Then it will be determined whether site-specific soil failures control the seismic HCLPF capacities of SSCs associated with the seismic accident sequences. Based on this outcome, the sequence-and plant-level seismic HCLPF capacity will be determined. Section 19.9.4 of the COL application will be revised to reflect this action. In a supplemental response to RAI 19-31 dated November 22, 2010 (ML103300212), the applicant provided the HCLPF capacities of site-specific SSCs: UHS/ Pumphouse/Cooling Tower structure, RSW Piping Tunnel, Diesel Generator Fuel Oil Storage Vault and Service Water Cooling Fans. The HCLPF capacities were calculated using the Conservative Deterministic Failure Margin (CDFM) method which is accepted in DC/COL-ISG-20 (ISG 20). It is shown that these HCLPF capacities are greater than 1.67 times the GMRS for STP 3&4. The staff found the applicant's response to RAI 19-31 adequate and acceptable. RAI 19-31 and Open Item 19-17 are considered to be resolved . The staff also confirmed that the proposed revision is incorporated into Revision 4 of the COL FSAR.

The staff reviewed the contents of Section 19.9 against ISG-20. The PRA-based SMA as accepted in NUREG–1503 for design certification generally meets the ISG-20. Since STP is referencing the certified design, the staff review focused on whether the provisions of ISG-20 in the COL stage (i.e., ISG-20 Section 5.2) are met. STP has committed to perform COL License Information Items 19.4 and 19.5 before the initial fuel loading. The response to RAI 19-7 stated that the site-specific GMRS is enveloped by the certified seismic design response spectra (CSDRS), the soil induced failures will be addressed, and the required seismic margins will be demonstrated.

The staff's evaluation according to ISG-20 concluded that site-specific effects are adequately considered and that the applicant's response to the COL license information items and the responses to other RAIs provide adequate confidence that the seismic fragility of SSCs and the plant level HCLPF will be maintained as stated in the design certification. The bases for the staff's conclusion are:

- Soil effects such as potential for soil liquefaction and slope failures are being addressed by STP per response to RAI 19-25.
- Site-specific structures (e.g., the UHS) were not modeled in the DC SMA. Therefore, the plant-level HCLPF will not be impacted by the fragilities of site-specific structures.
- Site-specific structures will be designed such that they will not collapse on or impact with other Seismic Category I structures modeled in the DC SMA.

Seismic Category I structures will be founded on soil with average shear wave velocities ranging from 776 ft/sec to 1000 ft/sec or on engineered structural fill. STP has committed to conduct site-specific soil-structure interaction (SSI) analysis, because the shear wave velocities are less than the 1000 ft/sec specified in the DC. Furthermore, the HCLPF capacities of SSCs shall be evaluated taking into account the site-specific effects and be provided before the initial fuel loading. (COM 19.9-4).

- COL License Information Item 19.5 Plant Walkdowns

In Section 19.9.5, the applicant commits (COM 19.9-5) to develop (before fuel loading) procedures for plant walkdowns to identify seismic, fire, and internal flooding vulnerabilities.

NRC staff found this commitment appropriate.

- COL License Information Item 19.6 Confirmation of Loss of AC Power Event

In FSAR Section 19.9.6, the applicant assesses site-specific parameters, such as specific causes of the LOOP, and their impact on a timely recovery of AC power. The NRC staff's review of this information is discussed in Section 19.3.4 of this SER.

- COL License Information Item 19.7 Procedures and Training for Use of AC Independent Water Addition

In Section 19.9.7, the applicant commits (COM 19.9-6) to develop and implement operating procedures and training for the ACIWA. These procedures will identify the system valve actuations, which provide the ACIWA via the RHR system as a water source to the RPV or to the containment.

NRC staff verified the flow path by checking Figures 5.4-10 and 9.5-4 and concluded that once developed and implemented, the operating procedures and training for these system valve actuations are reasonable.

- COL License Information Item 19.8 Actions to Avoid Common-Cause Failures in the Essential Communications Function (ECF) and Other Common-Cause Failures

In Section 19.9.8, the applicant commits (COM 19.9-7) to develop and implement (before fuel loading) test, maintenance, surveillance, and administrative procedures to ensure that credible common mode failures cannot occur. This commitment contains the provisions for procedure development that was described in the COL license information item in the DCD.

The staff reviewed the proposed commitment (including procedure development provisions) in the FSAR, and also examined the COL license information in the DCD. The staff found that the proposed commitment contains sufficient information for procedure development and is acceptable.

- COL License Information Item 19.9 Actions to Mitigate Station Blackout Events

In Section 19.9.9, the applicant commits (COM 19.9-8) to develop (before fuel loading) analyses and procedures to confirm the assumptions modeled in the PRA. Also, the PRA will be updated in accordance with 10 CFR 50.71(h)(1). This commitment contains the provisions for procedure development that was described in the COL license information item in the DCD.

The staff reviewed the proposed commitment (including procedure development provisions) in the FSAR, and also examined the COL license information item in the DCD. The staff found that the proposed commitment contains sufficient information for procedure development and is acceptable.

- COL License Information Item 19.10 Actions to Reduce Risk of Internal Flooding

In Section 19.9.10, the applicant commits (COM 19.9-9) to develop and implement (before fuel loading) training; design; and site-specific, PRA-based analyses and procedures to reduce the risk of internal flooding.

The text in Section 19.9.10, Item 8 (related to anti-siphon capability) is deleted to address Departure STP DEP 19R-1. In addition, Departure STP DEP 19R-1 addresses internal flooding of the control building due to the elimination of vacuum breaker valves on the supply and return piping, which connect to the RBCW heat exchangers. Elimination of the vacuum breaker valves is due to the RSW system design changes, including the use of horizontal-type pumps instead of vertical wet-pit type pumps and piping configuration changes between the UHS basin and the control building.

The deletion of text in Section 19.9.10, Item 8 appropriately reflects the RSW design changes under Departure STP DEP 19R-1, including the use of horizontal-type pumps instead of vertical wet-pit type pumps and piping configuration changes between the UHS basin and the control building. The impact of these RSW design changes on plant risk is evaluated in Appendix 19R of Chapter 19 of the STP Units 3 and 4 COL FSAR.

NRC staff determined that the supplemental information in Section 19.9.10 is also consistent with the internal flooding analysis developed in Appendix 19R. Based on this finding and the staff's safety evaluation of Appendix 19R associated with this probabilistic flooding analysis, the staff concluded that the supplemental information in Section 19.9.10 is acceptable.

- COL License Information Item 19.11 Actions to Avoid Loss of Decay Heat Removal and Minimize Shutdown Risk

In Section 19.9.11, the applicant commits (COM 19.9-10) to develop and implement (before fuel loading) operating procedures to avoid the loss of decay heat removal during a shutdown condition. The commitment contains the provisions for procedure development that was described in the COL license information item in the DCD.

The staff reviewed the proposed commitment (including procedure development provisions) in the FSAR, and also examined the COL license information in the DCD, as well as the evaluation of this COL action item in the ABWR DCD FSER. The staff found that the proposed commitment contains sufficient information for procedure development and is acceptable.

- COL License Information Item 19.12 Procedures for Operation of RCIC from Outside the Control Room

In Section 19.9.12, the applicant commits (COM 19.9-11) to develop procedures and conduct training for the RCIC operation. This commitment contains updated provisions for procedure development that was described in the COL license information item in the DCD.

The staff reviewed the proposed commitment (including updated procedure development provisions) in the FSAR, and also examined the COL license information in the DCD. The staff found that the proposed commitment contains sufficient information for procedure development and is acceptable.

- COL License Information Item 19.13 ECCS Test and Surveillance Intervals

In Section 19.9.13, the applicant commits (COM 19.9-12) to develop and implement (before fuel loading) a plan and procedures for identifying departures from the testing and surveillance intervals assumed in the PRA.

NRC staff determined that the supplemental information in Section 19.9.13 is appropriate and meets the objective of COL License Information Item 19.13.

- COL License Information Item 19.14 Accident Management

In Section 19.9.14, the applicant commits (COM 19.9-13) to include operator actions in the operating procedures and the training of these procedures be developed and implemented before fuel loading.

The human actions identified will be reviewed so that detailed procedures can be developed and the appropriate training will be conducted. These procedures will include the following:

  - Directions and guidance for operating the COPS shutoff valves. Appropriate care will be taken in the development of these procedures to ensure that the recovery of the containment heat removal or containment sprays does not induce late containment structural failure. If a suppression pool water level of at least 1 meter above the top of the highest horizontal connecting vent can be maintained following the COPS operation, the licensee may leave the shutoff valves open until after the recovery of containment heat removal, because the fission product release will be dominated by the initial noble gas release. In addition, the procedure for closing the shutoff valves will include steps for reintroducing nitrogen into the containment. When developing these accident mitigation strategies, the licensee will examine the potential benefits of the drywell spray operation if the containment fails in the drywell.

- For human actions to be taken that rely on instrumentation possibly operating outside of the qualification range, the licensee will determine the expected performance of the instrumentation and will provide additional guidance to the operator, if needed.

- Accident management strategies will consider the potential for recriticality during the recovery. A possible strategy could be a caution for the operators and/or technical support staff to monitor the power level (perhaps indirectly via the rate of containment pressurization) and enter procedures for anticipated transients without scram, as necessary.

NRC staff reviewed this information and determined that the information in the application does not address all of the items required to establish a sufficient technical basis for developing accident management procedures for STP Units 3 and 4. In particular, the licensee needs to further develop strategies for the containment to control the water level in the lower drywell and suppression pool after a vessel breach. Several candidate actions need to be addressed to minimize the release of radioactive materials into the environment and to achieve a safe and stable state, including the timely operation of the ACIWA, level control of the suppression pool and lower drywell using the ACIWA, and the cooling the upper drywell as much as possible after the containment has been vented through the COPS. For example, the premature actuation of the drywell flooder could lead to water being added into the lower drywell before a vessel breach. This could create the potential for a large ex-vessel steam explosion. The staff issued RAI 19-05 requesting the applicant to describe the necessary changes to the BWROG emergency procedure guidelines (EPGs) and severe accident guidelines (SAGs), as applied to the STP Units 3 and 4, to ensure sound and severe accident mitigation strategies and procedures.

The applicant's supplemental response to RAI 19-05 dated September 15, 2009 (ML092600154), states that the ABWR EPGs, which have been approved by the staff and incorporated by reference into the FSAR, were developed based on Revision 4 of the BWROG EPGs. During the process of reviewing the DCD for the ABWR, the staff evaluated major differences between the ABWR EPGs and Revision 4 of the BWROG EPGs. The results of this evaluation are documented in Section 18.8.5 of the ABWR FSER.

The applicant further states in the response to RAI 19-05 that it intends to follow NEI 91–04 Revision 1, "Severe Accident Closure Guidelines," which includes a commitment the industry made to the NRC to incorporate severe accident strategies into the overall Accident Management Program. Changes in the EPGs and SAGs (such as the containment flood strategy) will be included as inputs to the plant-specific technical guidelines. This is identified as Commitment Number COM 19.9-30 in U7-C-STP-NRC-100222, dated Ocotber 5, 2010 (ML102861292). The staff finds this approach acceptable, because it would utilize the technical basis for severe accident management procedures developed by STP.

Section 19.8.7 of the ABWR SSAR, discusses the ABWR features to mitigate severe accidents. Three of the features given in this section directly relate to the ABWR containment flooding strategy. These features include the RHR system, the ACIWA system, and the lower drywell flooder (LDF). Until the AC power is restored, keeping the COPS wetwell vent open is the only means of removing decay heat from the containment. The EPGs and SAGs must include a comprehensive strategy for utilizing all of the features related to the containment flooding strategies, minimizing releases of radioactivity, and maintaining a safe and stable state.

The technical basis for the ABWR EPGs was originally developed using MAAP-ABWR, which was a version of the MAAP3.0B code, modified to model the ABWR configuration. There were serious shortcomings in MAAP3.0B, so it was superseded by the MAAP4 code. The staff's comparative analyses have shown that significant differences in core melt progression can result, such that the technical basis for severe accident management must be changed in several respects. These changes need to be identified for the ABWR, and reflected in the ABWR EPGs and in equipment survivability determinations.

For example, the existing ABWR containment flood strategy emphasizes flooding the upper drywell to a level above the top of active fuel (TAF) to cool the debris in-vessel and prevent vessel breach. According to ABWR EPG Step C6-2, containment flooding would be terminated if, despite best efforts, the RPV level is below the TAF and the water level in the drywell has reached the bottom of the RPV. For this case, the staff wrote in the ABWR FSER (NUREG-1503) that the containment flood strategy is acceptable provided that the COPS is successfully actuated to relieve the pressure generated by an ex-vessel event that would lead to pressurization of the containment. Note, however, that the existing containment flood strategy does not address flooding in the lower drywell or controlling the suppression pool level with the ACIWA, by supplying fire water in either the drywell spray or wetwell spray mode. Additionally, there are no statements in the current ABWR EPGs about actions, equipment, and instrumentation to monitor and control the water levels in the suppression pool and the lower drywell.

Regarding the steam explosion potential from a premature opening of the drywell flooder, the applicant notes in the supplemental response to RAI 19-05 that high drywell gas temperatures are required to open up the flow paths from the suppression pool to the lower drywell, and these temperatures will occur after debris relocation from the vessel to the lower drywell. The staff's confirmatory assessment, however, indicates that lower drywell temperatures in some of the more likely severe accident scenarios may exceed 533 K (the temperature at which the fusible plugs will melt) before vessel breach. If this were the case, then molten core debris would fall into a water-filled lower drywell.

Even though the scenarios involved are highly unlikely, the staff believes that the existing containment flood guideline in the BRWOG's EPGs and SAGs may have to be revised to consider actions to address ex-vessel steam explosions. Since meeting COM 19.9-30 would address this concern, the staff considers that Open Item 19-5 is resolved.

- COL License Information Item 19.15 Manual Operation of MOVs

In Section 19.9.15, the applicant commits (COM 19.9-14) to develop and implement (before fuel loading) a procedure for operating MOVs manually.

NRC staff reviewed the proposed statements to develop and implement a procedure for manually operating the MOVs and found them reasonable and acceptable.

- COL License Information Item 19.16 High Pressure Core Flooder Discharge Valve

In Section 19.9.16, the applicant commits (COM 19.9 15) to develop and implement a procedure for verifying that the HPCF discharge valve is in the locked-open position before fuel loading.

NRC staff reviewed the proposed procedures and statements. The staff concluded that it is appropriate for the licensee to develop and implement a procedure for verifying that the HPCF discharge valve is in the locked-open position.

- COL License Information Item 19.17 Capability of Containment Isolation Valves

In Section 19.9.17 of FSAR Revision 3, the applicant committed (COM 19.9-16) to demonstrate that the stresses on the containment isolation valves will not exceed ASME Section III, Service Level C limits, and the ultimate pressure capability of the containment isolation valves will be greater than 1.03 MPa before fuel loading.

NRC staff issued RAI 19-32 asking the applicant to describe the method and tracking mechanisms to address this COL license information item. This RAI was tracked as Open Item 19-6 in the SER with open items.

The applicant's response to RAI 19-32 dated May 4, 2010 (ML101260119), provided a planned revision to FSAR Section 19.9.17 to address the design process for containment isolation valves and to discuss associated ITAAC. Subsequently, Revision 4 to the STP Units 3 and 4 FSAR states that containment isolation valves will be qualified by testing and analysis, and by satisfying the stress and deformation criteria at the critical locations within the valves. Per STP FSAR Revision 4, operability will be assured by meeting the requirements of the programs defined in Subsection 3.9.3.2, "Pump and Valve Operability Assurance," and Section 3.9.6, "Testing of Pumps and Valves," as supplemented in response to RAI 03.09.06-1 and in Sections 3.10 and 3.11. For containment isolation valves, STP FSAR Revision 4 states that ASME Code Certified Stress Reports will demonstrate that the stresses of the containment isolation valves, when subjected to the severe accident loadings of 0.77 MPa internal pressure and 260 °C (500 °F), in combination with dead loads, do not exceed ASME Section III, Service Level C limits. The individual parts of each valve will be verified not to exceed allowable structural capability limits under these severe accident conditions. In addition, the ASME Code Certified Stress Report will demonstrate the ultimate pressure capability at 260 °C (500 °F) to be at least 1.03 MPa. STP FSAR Revision 4 also states that acceptance criteria for ITAAC 2.14.1.2 will confirm the existence of an ASME Code Certified Stress Report for the containment pressure boundary components. This revision notes that the containment isolation valves are considered pressure boundary components and are included in the separate ASME Code Certified Stress Reports. The Certified Stress Reports for the containment isolation valves will include the stress analysis for the severe accident conditions of 0.77 MPa and 260 °C (500 °F). This revision also states that these actions will be completed before fuel loading, as part of Commitment COM 19.9-16. STP FSAR Revision 4 indicates that its provisions will be updated in accordance with 10 CFR 50.71(e) and based on the results of these analyses.

NRC staff found that Revision 4 to the STP Units 3 and 4 FSAR provides an acceptable description of the process to demonstrate the capability of the containment isolation valves. The description is consistent with the methodology specified in ABWR DCD Tier 2, Section 3.9. The ABWR ITAAC will provide confirmation of the completion of the design and qualification process for the containment isolation valves. Based on the applicant's planned FSAR revision dated May 4, 2010, NRC staff found that RAI 19-32 is resolved. Therefore, Open Item 19-6 is closed. The staff tracked this item as Confirmatory Item 19-16. The staff verified that the proposed revisions are in the COL FSAR Revision 4, and Confirmatory Item 19-16 is closed.

- COL License Information Item 19.18 Procedure to Ensure Sample Lines and Drywell Purge Lines Remain Closed During Operation

In Section 19.9.18, the applicant commits (COM 19.9-17) to develop operating procedures and administrative controls to ensure that sample lines and drywell purge lines remain closed during operation.

NRC staff reviewed the proposed statement and found this commitment appropriate.

- COL License Information Item 19.19 Procedures for Combustion Turbine Generator to Supply Power to Condensate and Condensate Booster Pumps

In Section 19.9.19, the applicant commits (COM 19.9-18) to develop and implement (before fuel loading) operating procedures for manually transferring the combustion turbine generator (CTG) power to the condensate, condensate booster pumps, and support systems.

NRC staff reviewed the proposed statement and found this commitment appropriate.

- COL License Information Item 19.19a Actions to Assure Reliability of the Supporting RCW and Service Water Systems

In Section 19.9.20, the applicant commits (COM 19.9-19) to develop and implement operating procedures for swapping the RCW and RSW operating pumps and heat exchangers at least monthly before fuel loading.

NRC staff reviewed the proposed statements to develop and implement a procedure. The staff concluded that it is appropriate for the licensee to develop and implement an operating procedure for swapping the RCW and RSW operating pumps and heat exchangers at least monthly.

- COL License Information Item 19.19b Housing of ACIWA Equipment

ABWR DCD, Revision 4, Section 19.9.21 states that if ACIWA equipment is housed in a separate building, that building must be capable of withstanding site-specific seismic events, flooding, and other site-specific external events such as high winds (e.g., hurricanes). The capability of the building housing the ACIWA equipment must be included in the plant-specific PRA. Accordingly, STP Units 3 and 4 COL FSAR Tier 2, Revision 4, Section 19.9.21 addresses the COL License Information item with a standard supplement and commitment (COM19.9-20) stating that the determination of the housing capability to withstand the site-specific seismic events, flooding, and other site-specific external events will be confirmed and will be included in the plant-specific PRA, which will be completed before fuel loading. NRC staff issued RAI 19-22 requesting the applicant to provide more detailed information addressing the approach, methods of analysis, computer codes, seismic structural modeling, damping, and pertinent sections of SRP acceptance criteria to be used in determining the housing structural capacity. The applicant's revised response to RAI 19-22 dated December 13, 2010 (ML103500240), describes in detail the location and the function of the ACIWA system, the analysis and design procedures, wind and seismic loadings, load combinations, codes and standards, SRP acceptance criteria, computer codes, and other design parameters to be used to evaluate the capability of the ACIWA housing to withstand the site-specific external events.

The staff's evaluation considered Table 19.8-2, "Important Features from Seismic Analyses," and ABWR DCD, Revision 4, which describes and lists the requirements for the ACIWA system as follows:

> Seismic qualification of the ACIWA system including the pumps, valves, and water supply [2.15.6 (SSE only)]. The collapse of the ACIWA building (shed) should not prevent the pumps from starting and running [2.15.6 (SSE only)]. All needed valves for system operation can be accessed and operated manually (2.15.6, 2.4.1). ACIWA can provide either vessel injection or drywell spray using equipment that does not require AC power. In addition, support systems normally required for ECCS operation are not required for ACIWA operation. ACIWA is an important system in preventing and mitigating severe accidents.

According to the above definition in the ABWR DCD, the ACIWA system is not a safety-related Seismic Category I system, but a system that is important in preventing and mitigating severe accidents. The ACIWA system is located in a separate building (together with the fire protection system) whose collapse should not prevent the ACIWA SSCs from performing their intended functions. The ACIWA housing is therefore a structure with Category II/I interaction potential that needs to be designed to comply with SRP 3.7.2.II.8. The staff was tracking RAI 19-22 as Open Item 19-16 in the SER with open items.

In the revised response to RAI 19-22 dated December 13, 2010 (ML103500240) the applicant states that the fire water pump house (FWPH) will be designed to meet the provisions of ASCE 7–05 using the design wind speed of 134 mph (3 second gust) with a return period of one in a hundred years and the site-specific SSE. The load combinations and acceptance criteria for ordinary commercial structures as specified in ASCE 7–05 will be adopted. The capability of the ACIWA housing against site-specific external events will be demonstrated as follows:

- The FWPH and the ACIWA equipment such as the direct diesel-driven pump and the associated piping and manual valves will be shown to have a seismic HCLPF of 0.5g peak ground acceleration.

- The FWPH is located above the design basis flood level of the site (MSL 33 ft) for non-safety-related structures, and therefore, flooding is not a design consideration for the non-safety-related ACIWA housing. In addition, in response to RAI 19-30 on July 28, 2010 (ML102110184) on external flooding, the status for all watertight doors and hatches was changed to be normally closed. This change in door status was documented in FSAR Section 2.4S.10, "Flooding Protection Requirements." Also, STP screened external flooding using the ASME/ANS RA-Sa-2009, Section 6-2.3, "The Fundamental Criteria for Screening External Events Other Than Fire and Seismic Events." Criterion (a) was used to screen external flood scenarios from detailed quantitative evaluation. Criterion (a) is satisfied since the STP design for safety-related systems, structures, and components satisfies the requirements of SRP Section 3.4.2, Revision 3, which was in effect at the time of the COL application.

- Tornados are not shown to be a significant risk contributor in ABWR DCD, Chapter 19, and accident sequences associated with tornado initiating events do not take credit for the ACIWA system.

- In response to RAI 19.01-31 dated February 16, 2011 (ML110490542), STP provided a sensitivity study to evaluate the effects of hurricane winds that exceed the STP Design Basis Wind Speed. The wind speed recurrence interval selected was 200 years. The ACIWA function and the CTGs are assumed to fail at this wind speed. Credit for the compensatory measures documented in FSAR Section 19.4.6, "ABWR Shutdown Risk" and

COM 19.4-1 yields a core damage frequency (less 1E-8/yr) which is significantly lower than the Commission goals for new reactors.

Therefore, the staff has found that the design of the ACIWA system meets Commitment 19.9-20 and the response to RAI 19-22 is therefore satisfactory. RAI 19-22 and Open Item 19-16 are resolved and closed.

- COL License Information Item 19.19c Procedures to Assure SRV Operability During Station Blackout

In Section 19.9.22, the applicant commits (COM 19.9-21) to develop and implement (before fuel loading) operating procedures to align stored nitrogen bottles for the SRVs.

NRC staff reviewed the proposed statement and found this commitment appropriate.

- COL License Information Item 19.19d Procedures for Ensuring Integrity of Freeze Seals

In Section 19.9.23, the applicant commits (COM 19.9-22) to develop and implement (before fuel loading) procedures for using and administratively controlling freeze seals.

NRC staff reviewed the proposed statement and found this commitment appropriate.

- COL License Information Item 19.19e Procedures for Controlling Combustibles During Shutdown

In Section 19.9.24, the applicant commits (COM 19.9-23) to develop and implement (before fuel loading) administrative procedures for controlling combustibles and ignition sources.

NRC staff reviewed the proposed statement and found this commitment appropriate.

- COL License Information Item 19.19f Outage Planning and Control

In Section 19.9.25, the applicant commits (COM 19.9-24) to develop and implement (before fuel loading) an outage planning and control program that is consistent with NUMARC 91-06 criteria.

NRC staff reviewed the proposed statement and found this commitment appropriate.

- COL License Information Item 19.19g Reactor Service Water Systems Definition

In Section 19.9.26, the applicant states that the overall results of the STP RSW evaluation are bounded by the conclusions of the standard ABWR DCD. The overall CDF increase due to the RSW/UHS design is small. NRC staff performed an audit and confirmed that the effect on CDF is small.

- COL License Information Item 19.19h Capability of Vacuum Breaker

In Section 19.9.27, the applicant commits (COM 19.9-25) to demonstrate (before fuel loading) the capability of the vacuum breaker seating material to withstand the temperature profiles associated with the equipment survivability requirements specified in Subsection 19E.2.1.2.3. As part of the commitment, the FSAR will be updated in accordance with 10 CFR 50.71(e) to reflect the results of this demonstration. The staff found this commitment acceptable.

- COL License Information Item 19.19i Capability of the Containment Atmospheric Monitoring System

In Section 19.9.28, the applicant commits (COM 19.9-26) to demonstrate (before fuel loading) that the containment atmospheric monitoring system can be exposed to containment pressure associated with the equipment survivability requirements specified in Subsection 19E.2.1.2.3. As part of the commitment, the FSAR will be updated in accordance with 10 CFR 50.71(e) to reflect the results of this demonstration. The staff found this commitment acceptable.

- COL License Information Item 19.19j Plant Specific Safety-Related Issues and Vendors Operating Guidance

In Section 19.9.29, the applicant commits (COM 19.9-27) to develop and implement (before fuel loading) plant operating procedures for maintaining the important safety functions during shutdown operations. The operating guidance from the vendors to perform control rod drives and reactor internal pump maintenance activities will also be implemented before fuel loading. The staff reviewed the proposed COL activities during shutdown in the DCD and the supplemental FSAR statement, as well as the evaluation of the COL activities in the ABWR DCD FSER. The staff found that the information is sufficient to accept the commitment.

- COL License Information Item 19.30 PRA Update

In Section 19.9.30, the applicant states that the standard PRA design was reviewed against site-specific design information (e.g., the UHS) and interface requirements of the standard design and was updated to ensure that the PRA results remain bounding. A delta-PRA was performed for those site characteristics that were not bounded by the PRA design results. The net impact of the STP-specific design shows a net decrease in risk compared to the standard ABWR PRA.

The staff also issued RAI 19.01-25 requesting the applicant to address how these commitments are being tracked. The applicant's response dated August 5, 2009 (ML092220163), states that Sections 19.9 and 19.4S of the DCD and FSAR include a number of commitments originating from the PRA. These commitments can be essentially grouped into:

- Develop EOPs and Abnormal Operating Procedures (AOPs),
- Develop procedures for performing a plant-specific PRA, and
- Develop other miscellaneous procedures relating to the PRA.

EOPs and AOPs will be verified and validated under the Human Factors Engineering Program and developed on a schedule to support the Plant Operations Training Program. Procedures for performing plant-specific PRA will be completed 1 year before fuel loading. The plant-specific PRA will be based on as-procured and as-built data and will be completed before fuel loading. Other miscellaneous procedures relating to the PRA will be completed 1 year before fuel loading.

The staff requested the applicant to provide more detailed information regarding the implementation schedules for the commitments in accordance with the guidance in RG 1.206, Section C.III.4.3 for COL license information items that will not be available prior to issuance of license. This issue was tracked as Open Item 19-7 (RAI 19.01-25) in the SER with open items. The applicant's supplemental response to RAI 19.01-25 dated January 14, 2010 (ML100190245), states that procedural requirements identified in "COL License Information" sections will be incorporated into the plant procedures. The milestones and program development plans are included in COL FSAR Section 13.5. For those "COL License Information Item" commitments that are related to new or conforming assessments, the FSAR will be updated in accordance with 10 CFR 50.71(e). The staff found this response acceptable, and issues related to “COL License Information” in Section 19.9 are resolved. The staff confirmed that the proposed revisions are incorporated into Chapter 19 of FSAR Revision 4.

19.9.5 Post Combined License Activities

The applicant identifies 27 commitments (COM 19.9-1 through 19.9-27) to be implemented in this section (see Section 19.9.4 above).

- In addition to the COL license information items in this section, there are other COL license information items in Section 19.4S and Appendices 19A and 19B. The staff issued RAI 19.01-25 asking the applicant to describe the plan and implementation schedules for these information items. With the discussion in Sections 19.4S.5 and 19.9.4, issues related to “COL License Information” in Sections 19.9 and 19.4S are resolved. The staff’s review of Appendices 19A and 19B are discussed in their corresponding sections of this SER.

19.9.6 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 NRC staff's review confirmed that the applicant has addressed the required information relating to “COL License 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 “COL License Information” that were incorporated by reference have been resolved.

In addition, based on the above discussion on the “COL License Information,” the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.10 Assumptions and Insights Related to Systems Outside of the ABWR Design (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.1.1, “Design Phase”; 19.1.2.1, “PRA Scope”; and 19.1.4.1.2, “Results from the Level 1 PRA for Operations at Power.”)

19.10.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.10 of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

19.10.2 Summary of Application

Section 19.10 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.10 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Section 19.10.1, the applicant provides the following:

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

- STP DEP 19R-1 — Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

This departure addresses the internal flooding of the control building due to the elimination of vacuum breaker valves on the supply and return piping connecting to the RBCW heat exchangers.

<u>Supplemental Information</u>

Section 19.10.1 Reactor Service Water (RSW) System and Safety-Related Ultimate Heat Sink (UHS) Assumptions

In this section, the applicant changes the assumptions that all RSW isolation valves receive an automatic close signal on a high water level in the control building RSW/RCW rooms.

19.10.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19.10.4 Technical Evaluation

NRC staff reviewed Section 19.10 of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL 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 "Assumptions and Insights Related to Systems Outside of the ABWR Design."

The staff reviewed the information in the COL FSAR:

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

- STP DEP 19R-1 — Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

The applicant deletes specific text in Section 19.10.1 related to "Anti-siphon Capability" to address Departure STP DEP 19R-1. These deletions do not affect the PRA, and therefore are acceptable.

The applicant evaluation in accordance with Item B.5 of Section VIII of Appendix A to 10 CFR Part 52 determined that the Tier 2 departures did not require prior NRC approval. Within the review scope of this section, the staff found it reasonable that this departure does not require prior NRC approval. The applicant process for evaluating departures and other changes to the DCD is subject to NRC inspections.

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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.

Supplemental Information

Section 19.10.1 Reactor Service Water (RSW) System and Safety-Related Ultimate Heat Sink (UHS) Assumptions

The applicant changed the PRA assumption that all RSW isolation valves receive an automatic close signal on a high water level in the control building RSW/RCW rooms. The applicant states that in each RSW division, there are redundant supply-side isolation valves that receive an automatic close signal on a high water level (1.5 meters) in the control building RSW/RCW room. This change in PRA assumption is consistent with the design departure (STP DEP 19R-1), and, therefore, NRC staff found this change acceptable.

19.10.5 Post Combined License Activities

There are no post COL activities related to this section.

19.10.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Assumptions and Insights Related to Systems Outside of the ABWR Design." 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 "Assumptions and Insights Related to Systems Outside of the ABWR Design" that were incorporated by reference have been resolved.

In addition, based on the above discussion on "Assumptions and Insights Related to Systems Outside of the ABWR Design," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.11 Human Action Overview (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.1.2.1, "Use of PRA in Support of Licensee Programs"; 19.1.3.4, "Use of the PRA in the Design Process"; 19.1.7.1, "PRA Input to Design Programs and Processes"; and 19.1.8, "Conclusions and Findings.")

19.11.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.11 of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

19.11.2 Summary of Application

Section 19.11 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.11 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Section 19.11, the applicant provides the following:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure addresses the issue that the pump and turbine are contained in same casing on a monoblock. The design eliminates many supporting components.

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

This departure eliminates obsolete data communication technology and the unnecessary and inadvertent actuation of prevention logic and equipment. The departure also changes the implementation, architecture, testing, and surveillance descriptions of the SSLC.

- STP DEP T1 5.0-1 Site Parameters

This departure addresses information pertaining to STP site parameters that are not bounded by those described in the ABWR DCD. A new human action is modeled by the STP Units 3 and 4 external flood analysis to close the control room watertight access door in the event of an external flood. This action is considered important and is discussed in Section 19R, "External Flooding."

Tier 2 Departure Requiring Prior NRC Approval

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

This departure changes the design to two MVES (13.4 kV and 4.6 kV) instead of the one 6.9 kV MVES described in the ABWR DCD.

The applicant has updated the importance of ranking Level 1 internal events, such as human-error probabilities, to reflect plant design changes for STP Units 3 and 4, site-specific characteristics and model enhancements. These changes do not modify the status of the four human actions to be taken after accident initiation. These actions are considered most important for the updated Level 1 internal event rankings.

19.11.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19.11.4 Technical Evaluation

NRC staff reviewed Section 19.11 of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic[1]. The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Human Action Overview."

The staff reviewed the information in the COL FSAR:

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 2.4-3 RCIC Turbine/Pump

The pump and turbine are contained in same casing on a monoblock; this design eliminates many supporting components. This departure does not affect the human error probability modeled in the STP site-specific PRA model.

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

This departure eliminates obsolete data communication technology and the unnecessary and inadvertent actuation of prevention logic and equipment. This departure does not affect the human error probability modeled in the STP site-specific PRA model.

- STP DEP T1 5.0-1 Site Parameters

STP site parameters are not bounded by those described in the ABWR DCD. A new human action is modeled by the STP Units 3 and 4 external flooding analysis to close the control room watertight access door in the event of an external flood. This action is considered important and is discussed in Section 19R, "External Flooding."

NRC staff determined that the departures under Section 19.9.3 appropriately reflect Departure STP DEP T1 5.0-1, as well as the departures related to the external flooding analysis under Appendix 19R. However, as a result of the open item identified under Appendix 19R that was associated with the external flooding analysis, the staff was unable to finalize the conclusions for these departures in Section 19.9.3. This issue was tracked as Open Item 19-12 (RAI 19-30) in the SER with open items. Based on the response to RAI 19-30 dated July 28, 2010 (ML102110184), the status of all watertight doors is normally closed. Therefore, no operator actions are required to implement flood protection measures as discussed in Section 2.4S.14 of the FSAR. Hence, the operator action previously described in this section of FSAR Revision 3 will be removed. Based on (1) the change in watertight door status to be normally closed, and (2) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-12 (RAI 19-30) have been resolved. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR. Therefore, the staff found the applicant's modeling of external floods acceptable.

Tier 2 Departure Requiring Prior NRC Approval

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

The ABWR Standard R-COL design modification states that a dual MVES consisting of 13.8 kV and 4.16 kV are used to replace the single 6.9 kV MVES in the ABWR DCD.

The evaluation of this departure is described in Section 19.3.4 of this SER.

19.11.5 Post Combined License Activities

- The applicant identifies commitment (COM 19.9-13) to address COL License Information Item 19.14 as discussed in SER Section 19.9.4.

19.11.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Human Action Overview." 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 "Human Action Overview" that were incorporated by reference have been resolved.

In addition, based on the above discussion on "Human Action Overview," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.12 Input to the Reliability Assurance Program (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.4.1.2, "Results from the Level 1 PRA for Operations at Power"; 19.1.4.2.2, "Results from the Level 2 PRA for Operations at Power"; 19.1.6.2, "Results from the Low-Power and Shutdown Operations PRA"; 19.1.7, "PRA-Related Input to Other Programs and Processes"; and 19.2.2, "Severe Accident Prevention.")

Section 19.12 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.12, "Input to the Reliability Assurance Program," of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding information outside of the DCD related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the "Input to the Reliability Assurance Program" have been resolved.

19.13 Summary of Insights Gained from the PRA (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.1.1, "Design Phase"; 19.1.2.1, "PRA Scope"; 19.1.4.1.2, "Results from the Level 1 PRA for Operations at Power"; and 19.2, "Severe Accident.")

19.13.1 Introduction

This section of the FSAR described the text changes and supplemental information in Section 19.13 of the U. S. ABWR DCD due to the departures of the STP Units 3 and 4 design from those described in the ABWR DCD.

19.13.2 Summary of Application

Section 19.13 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19.13 of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Section 19.13, 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 2.4-3 RCIC Turbine/Pump

This departure addresses the issue that the pump and turbine are contained in same casing on a monoblock. The design eliminates many supporting components.

- STP DEP T1 5.0-1 Site Parameters

This departure addresses information pertaining to STP site parameters that are not bounded by those described in the ABWR DCD.

19.13.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in section 19.1.3 of this SER.

19.13.4 Technical Evaluation

NRC staff reviewed Section 19.13 of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Summary of Insights Gained from the PRA."

The staff reviewed the information in the COL FSAR:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure deletes the RCIC lubricating oil cooling system from the text as a result of the new RCIC turbine/pump design. The ABWR DCD states that the RCIC lubricating oil cooling is mechanically driven by the turbine or pump shaft. Because of the new RCIC turbine/pump design, this statement is no longer applicable to the STP Units 3 and 4 FSAR. The applicant has also deleted this statement from the STP FSAR COL application. Therefore, the text changes in subsection 19.13.6.3 reflect the design departure.

- STP DEP T1 5.0-1 Site Parameters

Section 19.4 of the STP Units 3 and 4 FSAR discusses the impact of this departure on the external flooding analysis. To further reduce the susceptibility of an external flood, the applicant developed plant and site procedures. See Section 19.9.3 for a discussion of these procedures. NRC staff determined that the departures under Section 19.9.3 appropriately reflect Departure STP DEP T1 5.0-1, as well as the departures related to the external flooding analysis under Appendix 19R.

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.

19.13.5 Post Combined License Activities

The applicant identifies commitment (COM 19.9-13) to address COL License Information Item 19.14 as discussed in SER Section 19.9.4.

19.13.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Summary of Insights Gained from the PRA." 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 "Summary of Insights Gained from the PRA" that were incorporated by reference have been resolved.

In addition, based on the above discussion on "Summary of Insights Gained from the PRA," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19.14 <u>Loss of Large Areas of the Plant Due to Explosions or Fires</u>

19.14.1 Introduction

In a letter to the U.S. Nuclear Regulatory Commission (NRC), dated May 26, 2009, STP Nuclear Operating Company (STPNOC) submitted Revision 0 of the South Texas Project (STP) Units 3 and 4 Mitigative Strategies Report.

In the submittal, the applicant describes how the requirements to address loss of large areas (LOLAs) of the plant due to explosions or fires from a beyond-design basis event (BDBE) are met. These requirements are in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.80(d) and 10 CFR 50.54(hh)(2). It should be noted that the attachment to this safety evaluation (SE) section (Attachment A), as well as some documents referenced in this SE section, include security-related or safeguards information, and are not publicly available.

The provisions of 10 CFR 52.80(d) require an applicant for a combined operating license (COL) to submit a description and plans for implementation of the guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities under the circumstances associated with the LOLAs of the plant due to explosions or fire as required by 10 CFR 50.54(hh)(2).

The provisions of 10 CFR 50.54(hh)(2) require licensees to develop and implement guidance and strategies for addressing the LOLAs of the plant due to explosions or fires from a BDBE. Specifically, guidance and strategies are intended to maintain or restore core cooling, containment, and SFP cooling capabilities including:

- fire fighting
- operations to mitigate fuel damage
- actions to minimize radiological release

19.14.2 Summary of Application

In a letter dated May 26, 2009 (not publically available), the applicant for the STP COL application submitted its "Mitigative Strategies Report." The applicant will incorporate this report, including any applicable changes identified in response to NRC requests for additional information (RAIs), into a future revision to Part 11 of the STP COL application. The applicant stated that the LOLA mitigative strategies, including implementation of operational and programmatic aspects of responding to loss of large area events, would be implemented prior to initial fuel load.

19.14.3 Regulatory Basis

The applicable regulatory requirements for loss of large areas of the plant due to explosions or fires are as follows:

- 10 CFR 50.54(hh)(2)
- 10 CFR 52.80(d)

The applicable regulatory guidance includes Interim Staff Guidance (ISG) DC/COL-ISG-016, "Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d) Loss of Large Areas of the Plant due to Explosions or Fires from a Beyond-Design Basis Event" (not publically available), which provides an acceptable means of meeting the requirements of 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d). The DC/COL-ISG-016 references the February 25, 2005, guidance letter (not publically available) to operating reactor licensees for Phase 1 and the Nuclear Energy Institute (NEI) document NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline," Revision 3, for Phases 2 and 3 (not publically available). The DC/COL-ISG-016 takes exception to a few areas of NEI 06-12, and provides additional clarification and enhancement of NEI 06-12 and the staff's guidance letter issued February 25, 2005, based on NRC inspections of operating reactor implementation. The DC/COL-ISG-016 has two attachments: Attachment 1 is titled, "Supplementary Guidance for Implementing Mitigation Strategies," and Attachment 2 is titled, "Experience Gained from Implementation of Temporary Instruction 2515/171 at Currently Licensed Power Reactor Sites and Related Staff Positions."

19.14.4 Technical Evaluation

The staff reviewed the applicant's submittal consistent with the requirements of 10 CFR 52.80(d) and 10 CFR 50.54(hh)(2). The staff also used the guidance in DC/COL-ISG-016 to perform its review. The DC/COL-ISG-016 references the February 25, 2005, guidance letter for Phase 1, and NEI 06-12 for Phases 2 and 3. A further discussion of the staff's technical evaluation of the STP Units 3 and 4 submittal is found in Attachment A (non-public) to this Chapter 19 SER.

The STP COL applicant provided the LOLA event evaluation via a three-phased approach similar to existing plants and consistent with the NEI 06-12 guidance, Phases 1, 2, and 3. The applicant's MSR, dated May 26, 2009, was written at the programmatic level for licensing approval, and the implementation details and documentation will be made available for inspection by the NRC prior to initial fuel load. In response to NRC staff RAIs, the applicant submitted additional information to clarify the MSR. The applicant's responses to these RAIs are evaluated by the NRC staff in Attachment A (non-public) to Chapter 19 of this SER.

In its submittal of the MSR, the applicant provided a Mitigative Strategies Table (MST), which follows the template guidance in Appendix D to NEI 06-12. The MST addresses various areas and issues pertinent to loss of large areas and describes commitments for areas that are best resolved closer to the completion of construction STP Units 3 and 4. All commitments made in the submittal will be implemented prior to the initial fuel load of the units.

The MST addresses the three phases considered in NEI 06-12. The phases as described in the guidance documents can be mapped to the regulatory requirements and are as follows:

- Phase 1 – Fire Fighting Response Strategy
- Phase 2 – Spent Fuel Pool Cooling
- Phase 3 – Reactor Core Cooling and Fission Product Release Mitigation

Phases 1, 2, and 3 of NEI 06-12 are similar to the three areas included as part of the requirements in 10 CFR 50.54(hh)(2): fire fighting, operations to mitigate fuel damage, and actions to minimize radiological release. However, the three phases are categorized differently. In 10 CFR 50.54(hh)(2), the category of operations to mitigate fuel damage includes both the reactor core and the spent fuel pool, and the category of actions to minimize radiological release is separate. In NEI 06-12, spent fuel pool and reactor core cooling are found in separate phases, and reactor core cooling and fission product release mitigation are combined. Despite the change in the categorization of the phases in NEI 06-12 and the areas of the regulatory requirements, the staff finds all of the necessary information is included in the submittal.

The guidance for Phases 1, 2, and 3 suggests development of certain strategies or processes to mitigate the consequences of a LOLA event. The applicant addressed all of these suggested strategies or processes. In evaluating each plant specific mitigating strategy against its functional objective[1], the staff weighed whether the strategy reasonably can be expected to successfully provide spent fuel pool cooling, or maintain or restore the key safety functions necessary to protect the reactor core and containment. The staff's review considered the expected effectiveness of strategies and the ease and timeliness of strategy implementation.

While some strategies needed to meet 10 CFR 50.54(hh)(2) can be developed and implemented in the near future, some strategies and planning efforts cannot be effectively determined or implemented until the plant is further along in construction. To identify such commitments for future action, the applicant documented areas that would be more appropriately completed prior to the initial fuel load. The staff reviewed the commitments made by the applicant in its submittal and is satisfied that the timing of all procedural or strategy development was appropriately scheduled prior to the initial fuel load.

The MSR has been reviewed by the NRC staff for content using DC/COL-ISG-016, and found to include all strategies considered essential for such a program, and is acceptable. The staff finds that the regulatory requirements of 10 CFR 52.80(d) and 10 CFR 50.54(hh)(2) are met.

The NRC staff has identified as **Confirmatory Item 19.A-1** the revisions to Part 11 of the STP COL application to include the MSR proposed by the applicant in its October 4, 2010, letter, as modified in its letters dated January 5, 2011, February 2, 2011, February 14, 2011, and February 21, 2011. The specific modifications to the MSR are discussed in detail in Attachment A (non-public) to Chapter 19 of this SER.

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[1] As used here, the functional objective is the basic description of the capabilities of the conceptual strategy(s) as proposed for Phase 2 and 3 by NEI and accepted by NRC.

19.14.5 Post Combined License Activities

The staff proposes to include a license condition requiring the applicant to submit to the NRC an implementation schedule and updating it periodically for the strategies developed in accordance with 10 CFR 50.54(hh)(2) . In addition, the license condition will require the licensee to appropriately maintain those strategies.

19.14.6 Conclusion

The NRC staff reviewed the information provided by the applicant under 10 CFR 52.80(d), the staff concludes, pending closure of **Confirmatory Item 19.A-1**, that the applicant has adequately followed the guidance of DC/COL-ISG-016; NEI 06-12; and the February 25, 2005, guidance letter. The staff finds that the applicant provided sufficient information at the COL application stage, including commitments made in the STP COL application, to meet the requirements of 10 CFR 52.80(d) and to provide reasonable assurance that the requirements in 10 CFR 50.54(hh)(2) will be met prior to the initial fuel load of STP Units 3 and 4, respectively.

19A Response to CP/ML Rule 10 CDF 50.34(f) (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.2.6, "Consideration of Potential Design Improvements Under 10 CFR 50.34(f))."

19A.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19A of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

19A.2 Summary of Application

Appendix 19A of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19A of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19A, the applicant provides the following:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination (Table 19A-1)

This departure eliminates the hydrogen recombiner requirements.

.Supplemental Information

Section 19A.2.12 Evaluation of Alternative Hydrogen Control Systems (Item [1] [xii])

The ABWR primary containment is inerted and is therefore protected from hydrogen generation.

Subsection 6.2.7.1 for COL license information describes alternate hydrogen control. Section 6.2.5 describes the deletion of the flammability control system, including the recombiners, from the STP Units 3 and 4 design.

Section 19A.2.21 Hydrogen Control System Preliminary Design (Item [2] [ix])

The containment is inerted. See the response in Section 19A2.12.

Section 19A.2.46 Dedicated Penetration (Item [3][VI])

This item does not apply to the ABWR design.

<u>Section 19A.3 COL License Information Items</u>

The applicant included responses to the following COL license information items:

- COL License Information Item 19.20 Long-Term Training Upgrade
- COL License Information Item 19.21 Long-Term Program of Upgrading of Procedures
- COL License Information Item 19.22 Purge System Reliability
- COL License Information Item 19.23 Licensing Emergency Support Facility
- COL License Information Item 19.24 In-Plant Radiation Monitoring
- COL License Information Item 19.25 Feedback of Operating, Design and Construction Experience
- COL License Information Item 19.26 Organization and Staffing to Oversee Design and Construction
- COL License Information Item 19.27 Develop More Detailed QA Criteria

19A.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19A.4 Technical Evaluation

NRC staff reviewed Appendix 19A of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Response to CP/ML Rule 10 CFR 50.34(f)."

The staff reviewed the information in the COL FSAR:

<u>Tier 1 Departure</u>

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

Section 19E.4 of this SER evaluates 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.

Supplemental Information

19A.2.12 Evaluation of Alternative Hydrogen Control Systems (Item [1] [xii])

The ABWR primary containment is inerted and is therefore protected from hydrogen generation. Increasing the amounts of hydrogen moves the primary containment oxygen concentration further from the flammable regime. Radiolysis is the only potential source of oxygen in the ABWR primary containment.

Subsection 6.2.7.1 on COL license information describes alternate hydrogen control. Section 6.2.5 describes the deletion of the flammability control system, including the recombiners, from the STP Units 3 and 4 design and the design's capability to accommodate oxygen from radiolysis.

The staff agrees with the deletion of the texts in this Section of the FSAR.

19A.2.2 Hydrogen Control System Preliminary Design (Item [2] [ix])

The containment is inerted. See the response in Section 19A2.12 of this SER.

The staff agrees with the modified text in this Section of the FSAR.

19A.2.46 Dedicated Penetration (Item [3][VI])

This item does not apply to the ABWR design because the design has no external hydrogen recombiners. The staff agrees with this statement.

19A.3 COL License Information Items

- COL License Information Item 19.20 Long-Term Training Upgrade

STP Units 3 and 4 will include simulation facilities in accordance with 10 CFR 55.46 requirements for operator testing and licensing. Long-term operator training is addressed in Sections 18.8.and 13.2 of this SER.

- COL License Information Item 19.21 Long-Term Program of Upgrading of Procedures

Section 13.5 describes a long-term program of upgrading procedures for integrating and expanding efforts to improve plant procedures. The scope of the program includes emergency procedures; reliability analysis; human factors engineering; crisis management; operator training; and important industry, operation, and experience. This program is addressed in Section 13.5 of this SER.

- COL License Information Item 19.22 Purge System Reliability

Section 3.9 and Subsection 6.6.9.1 describe a testing program to ensure that the large ventilation valves close within limits that are assured in the radiologic design bases. This is addressed in Chapters 3 and 6 of this SER.

- COL License Information Item 19.23 Licensing Emergency Support Facility

Part 5 of this application provides a comprehensive site Emergency Plan that includes a description of the Emergency Operations Facility for STP Units 3 and 4. This is addressed in Section 13.3 of this SER.

- COL License Information Item 19.24 In-Plant Radiation Monitoring

Section 12.5.2 and Subsections 12.5.3.1 and 12.3.5.2 discuss personal monitoring and portable instrumentation of in-plant radiation and airborne radioactivity, as well as training and procedures appropriate for a broad range of routine and accident conditions. This is addressed in Chapter 12 of this SER.

- COL License Information Item 19.25 Feedback of Operating, Design and Construction Experience

This COL license information item addresses administrative procedures for evaluating operation, design, and construction experience and for ensuring that applicable and important industry experiences shall be provided in a timely manner to those designing and constructing the ABWR standard plant. Operator experience will be incorporated into training and procedures before fuel loading, as described in Sections 13.2.3 and 13.5.3, respectively. (COM 19A-1). This is addressed in Chapter 13 of this SER.

- COL License Information Item 19.26 Organization and Staffing to Oversee Design and Construction

Section 13.1 describes organization and staffing. This is addressed in Chapter 13 of this SER.

- COL License Information Item 19.27 Develop More Detailed QA Criteria

The Quality Assurance (QA) Program description is a separate document titled, "STP Units 3 and 4 Quality Assurance Program Description." This is addressed in Chapter 17 of this SER.

In RAI 19.01-25, the staff requested the applicant to provide more detailed information regarding the implementation schedules for the commitments in accordance with the guidance in RG 1.206, Section C.III.4.3, for COL license information that will not be available before issuance of the license. This RAI was tracked as open item 19-7 in the SER with open items. The applicant's supplemental response to RAI 19.01-25 dated May 19, 2010 (ML101410206), states that procedural requirements identified in the "COL License Information" will be incorporated into the plant procedures. The milestones and program development plans are included in COL FSAR Section 13.5. The COL FSAR will be revised as shown in the response to incorporate additional information on the COL license information items in Appendix 19A. The staff found this response acceptable because it resolves the issues related to "COL License Information" in Appendix 19A. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR.

19A.5 Post Combined License Activities

The applicant identifies the following commitment:

- Commitment (COM 19A-1) – The development and implementation of administrative procedures for evaluating operation, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the ABWR standard plant.

19A.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to “Response to CP/ML Rule 10 CDF 50.34(f).” 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 “Response to CP/ML Rule 10 CFR 50.34(f)” that were incorporated by reference have been resolved.

In addition, based on the above discussion on the “Response to CP/ML Rule 10 CFR 50.34(f),” the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19B Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.3.4, “Use of the PRA in the Design Phase.”)

19B.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19B of ABWR DCD due to the departures of the STP Units 3 and 4 design from those described in the ABWR DCD.

19B.2 Summary of Application

Appendix 19B of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19B of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19B, the applicant provides the following:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This departure addresses the elimination of the hydrogen recombiner requirements.

19B.2.18 A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

This departure revises the above ABWR DCD section to indicate that an inerted containment is used as a hydrogen control measure, and the applicant updates the 10 CFR 50.44 issuing date.

Tier 2 Departure Requiring Prior NRC Approval

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

This departure addresses the design change to utilize two MVES (13.4 kV and 4.6 kV) instead of the one 6.9 kV MVES described in ABWR DCD.

19B.2.11 A-35 Adequacy of Offsite Power System

The ABWR onsite power systems were to include three redundant and independent 6.9 kV class 1E safety buses. With this departure, the STP Units 3 and 4 onsite power systems include three redundant and independent 4.16 kV class 1E safety buses.

19B.3.1 COL Applicant Safety Issues

- COL License Information Item 19.28 COL Applicant Safety Issues

The applicant states that COL FSAR Section 1.9S addresses all COL issues related to Appendix 19B.

19B.3.2 Testing of Isolators

- COL License Information Item 19.28a Testing of Isolators

The applicant commits to develop an inspection and testing program for fiber optic-type isolators used between safety-related and nonsafety-related systems before fuel loading. (COM 19B-1).

19B.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19B.4 Technical Evaluation

NRC staff reviewed Appendix 19B of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic[1]. The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues."

The staff reviewed the information in the COL FSAR:

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

19B.2.18 A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

An inerted containment is used as a hydrogen control measure. This departure deletes the following words used in the acceptance criteria: "the provision for permanently installed hydrogen recombiners."

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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 the resolution section, the applicant updates the 10 CFR 50.44 issuing date from December 2, 1981, to September 16, 2003, for the latest revision. This minor change is corrected in the text. The staff found this change acceptable.

Tier 2 Departure Requiring Prior NRC Approval

19B.2.11 A-35 Adequacy of Offsite Power System

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

The ABWR onsite power systems were to include three redundant and independent 6.9 kV class 1E safety buses. With this standard departure, the STP onsite power systems now include three redundant and independent 4.16 kV class 1E safety buses. This change is reflected in the text. This departure's impact on the PRA is discussed in Section 19.3.4 of this SER.

COL License Information Items

- COL License Information Item 19.28 COL Applicant Safety Issues

Section 1.9S of the COL FSAR summarizes the resolution of generic issues and unresolved safety issues related to Appendix 19B. See SER Section 1.9 for further details.

- COL License Information Item 19.28a Testing of Isolators

The applicant commits to develop an inspection and testing program for fiber optic-type isolators used between safety-related and non-safety-related systems before fuel loading. (COM 19B-1).

The staff issued RAI 19.01-25 asking the applicant to describe the plan and implement schedules of these information items. The RAI was being tracked as Open Item 19-7.

The applicant's supplemental responses to RAI 19.01-25 dated May 19, 2010 (ML101410206), and August 18, 2010 (ML102320578), identify the updates of Generic Issues identified in Section 1.9S. The COL FSAR was to be revised as shown in the responses to incorporate additional information on the COL license information items in Appendix 19B. The staff found these responses acceptable because they resolve the issues related to "COL License Information" in Appendix 19B. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR.

19B.5 Post Combined License Activities

The applicant identifies the following commitment:

- Commitment (COM 19B-1) – The required testing, inspection, and replacement guidance will be developed and implemented before fuel loading. The applicant identifies this commitment to address COL License Information Item 19.28a.

19B.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues." 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 "Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues" that were incorporated by reference have been resolved.

In addition, based on the above discussion on the "Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19C Design Considerations Reducing Sabotage Risk

Section 19C of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19C, "Design Considerations Reducing Sabotage Risk," of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding information outside of the DCD related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to "Design Considerations Reducing Sabotage Risk" have been resolved.

19D Probabilistic Evaluations

Section 19D of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19D, "Probabilistic Evaluations," of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding information outside of the DCD related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to "Probabilistic Evaluations" have been resolved.

19E Deterministic Evaluations (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.3.3, "Design/Operational Features for Mitigating the Consequences of Releases from Containment"; 19.1.4.1.1, "Description of the Level 1 PRA for Operations at Power"; 19.1.4.3.1, "Description of the Level 3 PRA for Operations at Power (optional)"; 19.1.4.3.2, "Results from the Level 3 PRA for Operations at Power (optional)"; 19.2, "Severe Accident Evaluation"; 19.2.2, "Severe Accident Prevention"; 19.2.3, "Severe Accident Mitigation"; and 19.2.5, "Accident Management.")

19E.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19E of the U. S. ABWR DCD due to the departures of the STP Units 3 and 4 design from those described in the ABWR DCD.

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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.

19E.2 Summary of Application

Appendix 19E of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19E of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19E, the applicant provides the following:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This departure eliminates the hydrogen recombiner requirements.

Tier 2 Departures Not Requiring Prior NRC Approval

- STP DEP 2.2-5 CRAC2 and MACCS2 Codes

This departure uses the MACCS2 code for the offsite consequence analysis, thus replacing the CRAC2 code used in the ABWR DCD.

Also, this change to the MACCS2 code revises the descriptions in various subsections in this appendix.

- STD DEP 9.5-2 Lower Drywell Flooder Fusible Plug Valve

This departure replaces the fusible plug in the ABWR DCD design with a newer, temperature-sensitive fusible plug that melts at a specified temperature and, in turn, triggers the fusible plug valve to fully open.

Also, there are text revisions to the lower drywell flooder fusible plug valve description and opening time.

- STD DEP Admin (Table 19E.3-6, Case 5)

This departure corrects a typographical error in Table 19E.3-6.

19E.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19E.4 Technical Evaluation

NRC staff reviewed Appendix 19E of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL 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 "Deterministic Evaluations."

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

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

In the certified ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A, there is a flammability control system (FCS) consisting of two permanently installed, safety-related thermal hydrogen recombiners with associated piping, valves, controls, and instrumentation. The FCS was designed to control a potential buildup of hydrogen and oxygen in the containment from the radiolysis of water after a postulated design-basis LOCA. The staff found that this hydrogen release is not risk significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threathen containment integrity.

The NRC revised 10 CFR 50.44 to amend its standards for combustible gas control in light-water-cooled power reactors. As described in 10 CFR 50.44(c), for licenses issued after 2003, all containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction. The amended rule eliminates the requirements for hydrogen recombiners and relaxes the requirements for monitoring hydrogen and oxygen. The hydrogen/oxygen analyzers are maintained but as nonsafety-related. In STP COL Table 19.2-2, the applicant assesses no effect on the PRA because the recombiners are not modeled.

NRC staff reviewed STD DEP T1 2.14-1. The staff concurred that this change has no impact on the risk from severe accidents initiated during full power operation or on accident management strategies. During full power operation, the containment atmosphere is required by Technical Specifications to be inerted. The staff does, however, have concerns during startup and shutdown operations, when the containment would not be inerted.

Accordingly, the staff issued RAI 19-3, which asked the applicant to explain whether or not deleting the FCS, including the recombiners, affects the consideration of hydrogen combustion when the containment may not be inerted. RAI 19-3 also requested a discussion of the impacts on the large release frequency (LRF) and conditional containment failure probability (CCFP) from low-power and shutdown scenarios for STP Units 3 and 4. Subsequently, the staff issued RAI 19.01-31, related to Departure STD DEP 1.1-2, requesting the applicant to provide the shutdown and full power hurricane CDF and LRF, considering the shared fire water system. The staff also requested a description of the dominant sequences contributing to the shutdown and full power hurricane CDF and LRF estimates.

The applicant’s final response to RAI 19-3 dated January 20, 2010 (ML100250138), addressed the question related to removing hydrogen recombiners. The applicant made the following points pertaining to the risk significance from major hydrogen combustion during any severe accident that could be initiated during startup and shutdown operations.

- For severe accidents from full power, the hydrogen generation rate is sufficiently large that the H_2 recombiner is not effective in removing H_2 fast enough to prevent a deflagration in a deinerted containment. Inerting is an effective hydrogen control system for all risk-significant degraded core and full core melt accidents in these containments.

- For severe accidents from low power or shutdown with the containment intact, the hydrogen production would be similar to that from full power, although the time to reach the critical hydrogen production rate may be longer (i.e., longer time available for recovery).

  Because small differences in long duration recovery actions are not well characterized using available data or HRA methods, it is judged that the differences between effects of an at-power accident response or a shutdown accident response are not distinguishable.

- For severe accidents from low power without the containment intact, the H_2 recombiners are not effective and their presence is moot.

  During typical refueling operations, the containment would be deinerted and opened within the first 24 hours.

The applicant acknowledges the importance of defense-in-depth by pointing out the following pertaining to Low Power Shutdown (LPSD) events in the revised response:

An inerted containment atmosphere ensures there is insufficient oxygen to burn with any hydrogen in the containment. When inerting systems are unavailable or incapable of controlling combustible gas concentrations, the decisions and actions governing operation of drywell and suppression pool sprays provide a strategy to mitigate the consequences of a hydrogen generation event. Spray operation:

- Reduces containment pressure
- Reduces the flammability of combustible gases through the addition of water vapor to the gas mixture
- Suppresses the temperature and pressure increase following combustion if a deflagration does occur
  Scrubs the containment atmosphere in anticipation of radioactivity release
- Mixes the containment atmosphere to reduce localized buildup of combustible gases

Successful spray operation may also prevent containment venting at rates beyond allowable offsite radioactivity release rate limits for combustible gas control or delay its requirement until systems designed to control combustible gas concentrations can be restored to service. Therefore, the BWROG EPGs provide the defense-in-depth procedures to cope with combustible gas mixtures when the containment is deinerted. The actions to be taken by the crew include use of drywell sprays and judicious containment purge and vent operation. Removal of the hydrogen recombiners has no adverse effect on the availability of other systems included in plant-specific SAMGs for combustible gas control. If the reactor is shut down and deinerted, the applicant states that:

- There is a longer allowable time to take mitigative actions because the decay heat is lower than for full-power conditions.
- Recombiners would be ineffective because the containment is open to the environment.

- The primary method of combustible gas control under these conditions is the purging and venting of any gases from the containment to the environment via the Secondary Containment using normal ventilation systems.

In conclusion, the staff agrees with the applicant that the hydrogen recombiners would be ineffective for low power and shutdown severe accidents with the containment de-inerted. This conclusion applies for both conditions when the containment is isolated and is not isolated. For low power and shutdown conditions where the containment is de-inerted and containment not isolated, the staff agrees that the hydrogen recombiners are ineffective in treating the risk dominant severe accident challenges of hydrogen production. However, for this RCOL review, the staff does not assume that the risk associated with these conditions is very low since a shutdown level 1 and level 2 PRA was not performed for ABWR design certification.

It is important to note that a risk evaluation of design departures and site specific design features was performed as required by 10CFR52.79(d)(1) during low power and shutdown conditions. If applicable, the results of these evaluations were compared against the Commission's goals for new reactors. For these risk evaluations, the large release frequency was assumed to be approximately equal to core damage frequency. Additional information regarding low power and shutdown operation is discussed in Sections 19L and 19Q of this SER.

In the second part of RAI 19-3, the applicant was asked to provide a discussion of the impacts on the LRF or CCFP for low-power and shutdown core damage accidents. The staff realized that the response to RAI 19.01-31, which was being tracked as **Open Item 19-9**, would also address the concerns of the second part of RAI 19-3. Because Open Item 19-9 is now resolved and is being tracked as **Confirmatory Item 19-15** (see Appendix 19L below), RAI 19-3 is also resolved.

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

- STD DEP 2.2-5 CRAC2 and MACCS2 Code

In this departure, the applicant states that the STP COL FSAR evaluation of the consequences from potential radioactive releases used the MACCS2 computer code for the STP site. For the same potential scenarios, the ABWR DCD used the CRAC2 computer code for five sites, which are representative of each major geographical area of the United States and are described in detail in the certified ABWR DCD, Revision 4, Section 19E.3.

Section 19E.2 provides supplemental information detailing the various input parameters that were used. The applicant adds MACCS2 information to the various STP COL subsections, while retaining the ABWR DCD CRAC2 input information for comparisons and bounding evaluations. The applicant compares the site-specific results for three goals. Two goals are based on the NRC safety goal policy of minimizing risk to an individual and to the public near a plant. The third goal is industry-based and seeks to minimize the dose close to the plant. The results of this study show that (1) STP Units 3 and 4 satisfy these goals, and (2) the results of the ABWR DCD analysis using the CRAC2 code are bounding.

NRC staff considers the MACCS2 code to be an acceptable code for consequence analyses and therefore found the applicant's approach acceptable. The staff reviewed the site-specific inputs to the offsite, MACCS2 consequence analyses for potential severe accidents in Tables 19E.3-2 through 19E.3-4, 19E.3-6, and 19E.3-8 through 19E.3-13, included under Section 19.E of the STP Units 3 and 4 COL FSAR. Except for the release fractions in

Table 19E.3-6, the reported input quantities are reasonable and complete. Release fractions are only reported for three fission product groups: noble gases, iodine, and cesium. The applicant states that the remaining groups had negligible releases. However, the assessment of severe accident mitigation alternatives (SAMA) requires the consideration of all releases. Accordingly, the staff issued RAI 19-4 requesting the applicant to provide the complete list of release fractions for all cases that were evaluated. The applicant's response carries out an additional analysis that uses very conservative values for the releases being tracked by the additional fission product groups. The averted dose and cost risks increase slightly, but not enough to affect the SAMA evaluations. The staff found this reanalysis acceptable. Therefore, RAI 19-4 is resolved.

- STD DEP 9.5-2 Lower Drywell Flooder Fusible Plug Valve

This departure replaces the fusible plug in the ABWR DCD design with a newer, temperature-sensitive fusible plug that melts at a specified temperature and, in turn, triggers the fusible plug valve to fully open. In addition, the applicant provides supplemental information on the lower drywell flooder fusible plug valve description and opening time.

NRC staff reviewed STP DEP 9.5-2 included under Sections 9.5.12 and 19.E of the STP Units 3 and 4 COL FSAR. The text changes indicate that the LDF consists of ten pipes that run from the vertical pedestal vents into the lower drywell. Each pipe has an isolation valve and a fusible plug valve connected to the end of the pipe that extends into the lower drywell. The fusible plugs will melt when the surrounding air reaches a temperature of 533 °K (500 °F), after molten core debris enters the lower drywell. The fusible plug valve will open and will remain open to allow water to flow through each flooder pipe into the lower drywell and cover the core debris. The staff agreed that this concept would most certainly provide water to cover the debris. But the staff was concerned that the containment liner failure may not be averted for 24 hours after core damage. Accordingly, the staff decided to carry out a confirmatory assessment. To facilitate this assessment, the staff issued RAIs 19-1 and 19-28 requesting the applicant to provide the results of MAAP 4.0.7 calculations for the more likely severe accident scenarios for STP Units 3 and 4. The applicant provided the necessary information in a timely fashion. While the confirmatory assessment was in progress, the staff identified it as Open Item 19-13 in the SER with open items.

The staff performed the confirmatory assessment using the MELCOR 1.8.6 and MAAP 4.0.7 computer codes for two of the representative scenarios analyzed in Section 19E.2 of the ABWR SSAR with a modified version of MAAP3.0B known as MAAP-ABWR. These include a loss of core cooling with vessel failure at high pressure (LCHP) and a large LOCA with failure of all core cooling (LBLC). In the SSAR, these scenarios contributed 27 percent and 0.2 percent to the CDF, respectively. The LCHP represented station blackout-initiated severe accidents and was selected to evaluate behavior in the lower drywell following a vessel failure at high pressure. A sensitivity study was also carried out to determine the effects of vessel depressurization using the ADS. This scenario is called the loss of core cooling at low pressure (LCLP) sequence in the ABWR SSAR and contributes 62 percent to the CDF. Despite its low frequency, the LBLC was included to compare the simulations of a low-pressure severe accident following a LOCA. In addition to the base-case scenarios, sensitivity studies were performed to evaluate the consequences of a failure of the lower drywell flooder. In addition, a thorough analysis was performed to show the effectiveness of the ACIWA system.

Key results for the LCHP scenario are shown in Table 1. It is clear that the MAAP3-ABWR results reported in the DCD are considerably more optimistic than those calculated using the more up-to-date MAAP 4.0.7 and MELCOR 1.8.6 codes. The time to wetwell vent opening is considerably longer, the in-vessel hydrogen production is considerably less, and the fission product releases are much less. Similar differences are also calculated for the other two scenarios. The MAAP 4.0.7 results are also closer to the MELCOR 1.8.6 results than they are to the MAAP3-ABWR results.

Table 1. Key Results for the LCHP Scenario

Key Accident Parameter	MELCOR 1.8.6	MAAP 4.0.7	MAAP3-ABWR (DCD)
Top of core uncovered, min.	~7	27	18
Relocation of fuel to lower head, hr.	3.04	2.71	2.0
Drywell sprays on, hr	6.0	6.0	4.0
RPV failure, hr.	6.14	4.61	2.0
Start lower drywell flooding	6.14	4.61	2.0
Wetwell vent opens, hr.	14.0	17.9	25.0
Lower drywell flooding ceases (level in suppression pool below bottom of top vent), hr	82	79	N/A
Dryout of LDW floor, hr	86	111	N/A
Concrete erosion and noncondensable gas release resume, hr	86	119	N/A
Total in-vessel hydrogen production, kg	1410	543	177
Mass fraction of Iodine released to environment			
2 days (20 hours for MAAP3-ABWR)	9E-3	8E-4	2E-7
4 days	N/A	9E-2	N/A
Mass fraction of Cesium released to environment			
2 days (20 hours for MAAP3-ABWR)	1E-3	9E-5	1E-5
4 days	5E-2	6E-3	N/A
Mass fraction of Barium released to environment			
2 days	4E-5	7E-6	N/A
4 days (6 days for MAAP 4.0.7)	3E-4	2E-4	N/A

For both base-case scenarios, MELCOR and MAAP4 simulations predicted that containment overpressure failure would be averted for at least 24 hours after core damage. Without the proper functioning of the lower drywell flooder (a very unlikely outcome), MAAP predicted that liner integrity would not be maintained for at least 24 hours after core damage. The MELCOR calculation, however, predicted a basemat melt-through somewhat later (within 48 hours).

The confirmatory assessments also show that as long as AC power is not recovered and no other water source is provided, the core debris would eventually boil away the water in the lower drywell. The water level in the suppression pool would also diminish and the upper vent would eventually be uncovered. Within four days after core damage, the core debris would become molten again and core debris-concrete interactions would resume, leading to large releases of hydrogen, carbon monoxide, and fission products into the containment. In addition, significant releases of volatile fission products into the environment are predicted to occur. To prevent this from happening, the ACIWA system would need to be implemented to provide water to the suppression pool and to the lower drywell in a controlled fashion, preferably within 10 to 12 hours but no more than two days after core damage.

The confirmatory assessments provide important insights for developing severe accident management guidance. In the calculations that credit the activation of the lower drywell flooder, the molten core debris is cooled below the ablation temperature (1450K for the basaltic concrete), but the cooling is quick and fully effective only for the LCHP scenario, wherein the erosion is stopped at just 5 cm. In the low-pressure LCLP scenario, erosion is stopped at 17 cm. In the base-case LBLC scenario, the erosion is never completely stopped, though it is greatly slowed starting 2 hours after vessel breach; at 48 hours the erosion depth reaches 1.84 m. For each of the scenarios, MAAP4 predicted very little erosion as long as water covered the debris.

The containment can fail when the drywell head is lifted (incipient lifting occurs at 4.6 bar) if the elastomer head seal has previously been heated to above 533K. This situation arises in the LBLC scenario but not in the two station blackout scenarios, wherein the upper drywell temperatures remain lower. When this mode of drywell failure occurs, it is the dominant radiological release path to the environment. However, because the discharge rates are small, this mechanism does not have a large direct effect on containment pressurization.

Activation of the lower drywell flooder prior to vessel breach has been considered in a sensitivity case for the LBLC scenario, since in the base case the required activation temperature is actually predicted to be just barely attained at about 7 hours before vessel breach. (Since the precise peak containment temperature at such times depends on in-vessel conditions which likely are relatively model-dependent, the base case discounted the premature flooder actuation, and assumed flooder actuation to occur just after vessel breach.) In consequence, in the sensitivity case, 494 tonnes of water are in the lower drywell at the time of vessel breach. Barring the unlikely possibility of a steam explosion, (see the discussion of COL Information Item 19.14 in Section 19.9.4), the premature flooding has little consequence on debris/concrete interactions, containment pressurization, or radiological releases.

Three variants of the LCHP scenario have investigated the consequences of realigning the drywell sprays source from the wetwell pool to externally-supplied firewater, assumed to occur 10 hours into the accident. It is found that an injection rate of 0.008 m^3/s is approximately optimal, in that it allows the level in the wetwell pool to stay roughly constant (neither uncovering the top vent nor flooding the vacuum breaker) for times as long as 100 hours or more. As long as this situation prevails the fission product releases remain low, since the suppression pool effectively scrubs aerosols. As discussed below, however, MELCOR predicts that, when the wetwell vent opens, considerable quantities of fission product vapors flow through the water, into the wetwell air space, and out the wetwell vent to the environment.

A higher injection rate of 0.04 m^3/s causes the wetwell pool to fill until the rising water blows open the wetwell rupture disk, occurring at about 47 hours according to MELCOR. Large fission product releases are then predicted because the water contains much radioactive material. Much of the radioactive water, however, should remain in the vent stack.

If no firewater is added, the wetwell level is predicted to fall until the water in the LDW becomes isolated from the suppression pool water (i.e., the connection afforded by the passive flooder becomes uncovered). Dryout of the LDW and resumption of concrete attack is then predicted to occur at 86 hours; large environmental releases follow, via a pathway that becomes available for fission products to be transported from the vessel to the wetwell vent and out to the environment. These releases are caused by revaporization of volatile fission products deposited in the vessel, leading to protracted releases days into the accident. These fission products flow out of the failed lower vessel head into the drywell, and then into the downcomers and through the uncovered vents into the wetwell. From there, they flow out of the wetwell vent into the environment. It is clear that, from an accident management perspective, it would be necessary to add firewater in a controlled manner, preferably by 10-12 hr after start of the accident.

MAAP4 generally predicts enhanced cooling of molten core debris over the drywell floor as compared to the MELCOR results. MELCOR predicts containment failure due to lifting of the drywell head by pressure during the LCLB scenario, with the elastomer seal having failed due to high temperature; in MAAP4, this failure is predicted not to occur. Otherwise, the agreement between the codes on thermal-hydraulic predictions is generally reasonable. However, MELCOR predicts higher radiological releases. MELCOR predicts significant amounts of radioactive vapors in the vessel starting soon after the first gap releases. MAAP4 does not predict such vapors, so the higher MELCOR releases can be attributed to the less effective scrubbing of fission product vapors by the suppression pool (relative to the scrubbing of aerosols).

Most of the MAAP4 calculations indicate that the cumulative environmental releases are still slowly increasing even at long times. These releases are caused by revaporization of volatile fission products deposited in the vessel, leading to protracted releases late into the accident. These fission products flow out of the failed lower vessel head into the drywell, and then into the downcomers and through the vents into the wetwell.

See the above discussion on “Accident Management” in Section 19.9.4 for further information.

The MELCOR simulation of the LBLC scenario identified a circumstance where the drywell flooder could open prematurely to deliver suppression pool water into the lower drywell before a vessel breach. This circumstance could create the potential for a large ex-vessel steam explosion. The MAAP calculation, however, did not produce the same result. Because this accident scenario has such a low probability of occurrence, the staff believes that the premature actuation of the lower drywell flooder is extremely unlikely.

With the confirmatory assessment now complete, the staff considers Open Item 19-13 to be resolved.

- STD DEP Admin

The applicant defines administrative departures as minor corrections, such as editorial or administrative errors in the referenced ABWR DCD (i.e., misspellings, incorrect references, table headings, etc.). This departure corrects a typographical error in Table 19E.3-6. The staff finds this Admin departure reasonable.

19E.5 Post Combined License Activities

The applicant identifies commitments (COM 19.9-1, COM 19.9-8, COM 19.9-13, COM 19.9-21, and COM 19.9-25) to address COL License Information Items 19.1, 19.9, 19.14, 19.19c, 19.19h, and 19.19i as discussed in SER Section 19.9.4.

The staff issued RAI 19.01-25 asking the applicant to describe the plan and implementation schedules of these information items. The staff reviewed the applicant's supplemental response to this RAI in Section 19.9 of this SER and found the response acceptable. Therefore, all issues related to the above commitments are resolved.

19E.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Deterministic Evaluations." 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 "Deterministic Evaluations" that were incorporated by reference have been resolved. In addition, based on the above discussion on "Deterministic Evaluations," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19EA <u>Direct Containment Heating</u>

Appendix 19EA of the STP COL FSAR incorporates by reference with no departures or supplements Appendix 19EA, "Direct Containment Heating," of Revision 4 of the ABWR DCD, which is incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this appendix remains for review. [1] The staff's review confirmed that there is no outstanding issue related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to "Direct Containment Heating" have been resolved.

19EB <u>Fuel Coolant Interactions</u>

Appendix 19EB of the STP COL FSAR incorporates by reference with no departures or supplements Appendix 19EB, "Fuel Coolant Interactions," of Revision 4 of the ABWR DCD, which is itself incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this appendix remains for review.[1] The staff's review confirmed that there is no outstanding issue related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A,

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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.

Section VI.B.1, all nuclear safety issues relating to “Fuel Coolant Interactions” have been resolved.

19EC Debris Coolability and Core Concrete Interaction

Appendix 19EC of the STP COL FSAR incorporates by reference with no departures or supplements Appendix 19EC, “Debris Coolability and Core Concrete Interaction,” of Revision 4 of the ABWR DCD, which is incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this appendix remains for review.[1] The staff’s review confirmed that there is no outstanding issue related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to “Debris Coolability and Core Concrete Interactions” have been resolved.

19ED Corium Shield

Appendix 19ED of the STP COL FSAR incorporates by reference with no departures or supplements Appendix 19ED, “Corium Shield,” of Revision 4 of the ABWR DCD, which is incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this appendix remains for review.[1] The staff’s review confirmed that there is no outstanding issue related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the “Corium Shield” have been resolved.

19EE Suppression Pool Bypass

Appendix 19EE of the STP COL FSAR incorporates by reference with no departures or supplements Appendix 19EE, “Suppression Pool Bypass,” of Revision 4 of the ABWR DCD, which is incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this appendix remains for review.[1] The staff’s review confirmed that there is no outstanding issue related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the “Suppression Pool Bypass” have been resolved.

19F Containment Ultimate Strength

Appendix 19F of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19F “Containment Ultimate Strength” of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A, with no departures or supplements. NRC staff reviewed the application and considered the referenced DCD to ensure that no issue relating to this appendix remains for review.[1] The staff’s review confirmed that there is no outstanding information outside of the DCD related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the containment ultimate strength have been resolved.

19FA Containment Ultimate Strength

Appendix 19FA of the STP COL FSAR incorporates by reference with no departures or supplements Appendix 19FA, “Containment Ultimate Strength,” of Revision 4 of the ABWR DCD, which is incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this appendix remains for review. [1] The staff's review confirmed that there is no outstanding issue related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the "Containment Ultimate Strength" have been resolved.

19G Not Used

This appendix is not used in both the ABWR DCD and the applicant's FSAR.

19H Seismic Capacity Analysis

19H.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19H of the U. S. ABWR DCD due to the departures of the STP Units 3 and 4 design from those described in the ABWR DCD.

19H.2 Summary of Application

Appendix 19H of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19H of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19H, the applicant provides the following:

Tier 1 Departure

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

This departure reclassifies the RWB as a non-seismic structure.

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP Admin

This departure addresses editorial/nomenclature changes in Table 10H-1.

19H.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19H.4 Technical Evaluation

NRC staff reviewed Section 19H of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL

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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

represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Seismic Capacity Analysis."

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

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

This departure deletes the description of the RWB as a Seismic Category I structure from the ABWR DCD. See Sections 19.4 and 3.8 of this SER for the NRC staff's evaluation.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP Admin

The applicant defines administrative departures as minor corrections, such as editorial or administrative errors in the referenced ABWR DCD (i.e., misspellings, incorrect references, table headings, etc.). This departure addresses editorial/nomenclature changes in Table 10H-1. The staff finds this Admin departure reasonable.

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

19H.5 Post Combined License Activities

There are no post COL activities related to this section.

19H.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 NRC staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to "Seismic Capacity Analysis" that were incorporated by reference have been resolved.

19I Seismic Margins Analysis

19I.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19I of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

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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.

19I.2 Summary of Application

Section 19I of the STP Units 3 and 4 COL FSAR incorporates by reference Section 19I of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19I, the applicant provides the following:

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 19I.7-1 Atmospheric Control System Bypass Analysis

This departure replaces the MOVs with air-operated valves.

- STP DEP 1.1-2 Dual Units at STP Units 3 & 4

This departure clarifies that a single fire protection system water volume is used for dual units.

19I.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19I.4 Technical Evaluation

NRC staff reviewed Appendix 19I of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Seismic Capacity Analysis."

The staff reviewed the information in the COL FSAR:

Tier 2 Departures Not Requiring Prior NRC Approval

- STP DEP 1.1-2 Dual Units at STP Units 3 & 4

In Section 19I.3.1, "Support State Event Tree," the applicant states that:

> The STP Units 3 and 4 ABWR dual unit design will use the same fire protection system water volume as the single unit design of the reference ABWR DCD as described in STP DEP 1.1-2. This aspect does not change the SMA conclusions that no HCLPF accident sequence is less than two times the SSE.

NRC staff evaluated the above assertion from the standpoint of seismic capacity/fragility and found the justification acceptable.

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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 19I.7-1 Atmospheric Control System Bypass Analysis

This departure changes the atmospheric control system crosstie to air-operated valves, which allows for remote operation in a seismic event. As indicated in Section 19I.7, "Containment Isolation and Bypass Analysis," the analysis in the STP Units 3 and 4 FSAR has been changed to reflect the design of air operators on these valves. As a result, the seismic-induced bypass analysis of these lines is the same as the analysis described for the drywell inerting/purge lines.

NRC staff concluded that changing the design input assumption used in the seismic margins PRA analysis, as it relates to the design of the ACS crosstie lines/valves, is a correction of the basis for the PRA analysis and has no effect on the plant design or safety analysis.

The applicant evaluation in accordance with Item B.5 of Section VIII of Appendix A to 10 CFR Part 52 determined that the Tier 2 departures did not require prior NRC approval. Within the review scope of this section, the staff found it reasonable that these departures do not require prior NRC approval. The applicant process for evaluating departures and other changes to the DCD is subject to NRC inspections.

19I.5 Post Combined License Activities

There are no post COL activities related to this section.

19I.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 NRC staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to "Seismic Margins Analysis" that were incorporated by reference have been resolved.

19J Not Used

This appendix is not used in both the ABWR DCD and the applicant's FSAR.

19K PRA-Based Reliability and Maintenance (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.4.1.2, "Results from the Level 1 PRA for Operations at Power"; 19.1.4.2.2, "Results from the Level 2 PRA for Operations at Power"; 19.1.6.2, "Results from the Low-Power and Shutdown Operation PRA"; 19.1.7.4, "RA Input to the Reliability Assurance Program"; and 19.2.2, "Severe Accident Prevention.")

19K.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19K of the U. S. ABWR DCD due to the departures of the South Texas Projects Unit 3 and 4 design from those described in the ABWR DCD.

19K.2 Summary of Application

Appendix 19K of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19K of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19K, the applicant provides the following:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure addresses the pump and turbine monoblock design (pump and turbine are contained in the same casing), which simplifies the design and removes multiple components.

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

This departure eliminates obsolete data communication technology and the unnecessary, inadvertent actuation of prevention logic and equipment. Clarifications in the I&C nomenclature reflect the changes in this departure.

- STP DEP T1 5.0-1 Site Parameters

This departure addresses the site design-basis flood level, the maximum design precipitation rate for rainfall, the humidity (represented by the wet-bulb temperature), and the shear wave velocity at the STP site that are not bounded by those parameters described in the ABWR DCD.

Tier 2 Departure Requiring Prior NRC Approval

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

This departure changes the design to utilize two MVES (13.4 kV and 4.6 kV) instead of the one 6.9 kV MVES described in the ABWR DCD.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 19.3-1 Evaluation of Common Cause Failures

The common cause factors were added to the ABWR plant model used to quantify the effects of plant-specific factors for STP Units 3 and 4. The addition of the common cause terms represents a departure from the PRA that is described in the reference DCD.

- STP DEP 19R-1 Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

This departure addresses the internal flooding of the control building due to the elimination of vacuum breaker valves on the supply and return piping that connects to the RBCW heat exchangers. The departure deletes the words “anti-siphon capability” because the RSW no longer requires that capability.

Supplemental Information

19K.3 Determination of “Important Structures, Systems and Components” for Level 1 *Analysis*

The STP PRA identifies 14 SSCs that have the greatest importance in modest values of Fussell-Vesely (FV) and nine additional SSCs with the modest values of risk achievement worth. SSAR Section 19D.7 addresses significant human errors. Important SSCs under consideration for periodic testing and/or preventive maintenance as part of the RAP are identified in Section 19K.11.

19K.7 Determination of "Important Structures, Systems and Components" for Flood Analysis

- The applicant provides site-specific supplemental information in Section 19K.7 of the STP Units 3 and 4 COL FSAR that identifies important SSCs in the probabilistic flooding analysis of the relocated RSW pump house.

- The applicant provides site-specific supplemental information in Section 19K.7 of the STP Units 3 and 4 COL FSAR that identifies important SSCs in the probabilistic flooding analysis for external flooding, which addresses departure STP DEP T1 5.0-1 ("Site Parameters").

19K.10 Identification of Important Capabilities Outside the Control Room

The applicant identifies the following additional important activity:

- Verifying that all watertight doors are closed upon notification of the main cooling reservoir breach.

19K.11.1 Component Inspections and Maintenance

The following additional STP SSCs also have a high FV importance:

> The RBCW and RSW systems have a high FV importance with respect to CCF impacts, because these systems support a number of front-line safety systems. There are maintenance and testing tasks for the key components in each division, including pumps, heat exchangers, and the service water cooling tower fans.

19K.11.13 Flood Protection

This section lists and describes the important SSCs for flood protection:

- Watertight doors on external entrances to the control and reactor buildings, including the watertight barriers on the equipment access to the diesel generator rooms and in the emergency core cooling systems (ECCS)

- RSW pump house, pump rooms, and other rooms

- RCW rooms

- RSW and CWS isolation valves

- Circuit breakers that trip the RSW pumps and water level sensors in the turbine building condenser pit

19K.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19K.4 Technical Evaluation

NRC staff reviewed Appendix 19K of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "PRA-Based Reliability and Maintenance."

The staff reviewed the information in the COL FSAR:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump
- STD DEP T1 3.4-1 Safety-Related I&C Architecture
- STP DEP T1 5.0-1 Site Parameters

The above departures are evaluated in other sections of this SER (e.g., Section 19.11.4) and will not be discussed here.

Tier 2 Departure Requiring Prior NRC Approval~~.~~

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

The above departure is evaluated in other sections of this SER (e.g., Section 19.3.4) and will not be discussed here.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 19.3-1 Evaluation of Common Cause Failures
- STP DEP 19R-1 Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

The above departures are evaluated in other sections of this SER (e.g., Section 19.3.4 and Appendix 19R) and will not be discussed here.

Supplemental Information

19K.7 Determination of "Important Structures, Systems and Components" for Flood Analysis

- The staff reviewed the supplemental information related to the identification of important SSCs in the probabilistic flooding analysis of the relocated RSW pump house included under Section 19K.7 of the STP Units 3 and 4 COL FSAR. The staff determined that Section 19K.7 sufficiently identifies the important SSCs in this probabilistic flooding analysis developed under Appendix 19R. Based on this finding and the staff's safety evaluation of Appendix 19R associated with this probabilistic flooding analysis, the staff concluded that the supplemental information in Section 19K.7 is acceptable.

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

- The staff reviewed the supplemental information related to the identification of important SSCs from the probabilistic flooding analysis for external flooding included under Section 19K.7 of the STP Units 3 and 4 COL FSAR. The staff determined that Section 19K.7 sufficiently identifies the important SSCs in this probabilistic flooding analysis developed under Appendix 19R. However, as a result of the open item identified under Appendix 19R that was associated with this probabilistic flooding analysis, the staff was unable to finalize the conclusions relating to the supplemental information in Section 19K.7 associated with the probabilistic flooding analysis for external flooding. The staff was tracking this issue as Open Item 19-12 (RAI 19-30) in the SER with open items. In response to RAI 19-30 dated July 28, 2010 (ML102110184) and as discussed in Section 19R of this SER, the applicant revised COL FSAR Section 2.4S.10, "Flooding Protection Requirements," to state that all watertight doors and hatches are normally closed. Based on (1) the change in the watertight door status to be normally closed, (2) the removal of the screening quantification for the postulated main cooling reservoir breach from the FSAR, and (3) the revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-12 (RAI 19-30) are resolved. Therefore, the staff found the applicant's modeling of external floods for shutdown and full power to be acceptable.

19K.11 Reliability and Maintenance Actions

NRC staff reviewed the supplemental information in FSAR Section 19K.11, as part of the review of FSAR Section 17.4S. The discussion of this review is in SER Section 17.4S.4.3. This review identified several confirmatory items related to FSAR Section 19K.11. FSAR Section 19K.11 is also dependent on the probabilistic external flooding analysis under Appendix 19R, in which the staff identified Open Item 19-12. As stated above, the applicant has revised the FSAR to change the watertight status to be normally closed. The staff's review found the applicant's modeling of external floods acceptable, and Open Item 19-12 is closed.

19K.5 Post Combined License Activities

There are no post COL activities related to this section.

19K.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "PRA-Based Reliability and Maintenance." With the exception of the confirmatory items identified in the discussion under SER Subsection 17.4S.4.3, 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 "PRA-Based Reliability and Maintenance" that were incorporated by reference have been resolved, In addition, based on the above discussion on "PRA-Based Reliability and Maintenance," the staff concluded, pending the completion of the above confirmatory items, that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19L ABWR Shutdown Risk Evaluation (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.6.1, "Description of the Low-Power and Shutdown Operations PRA.")

19L.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19L of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

19L.2 Summary of Application

Appendix 19L of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19L of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19L, the applicant provides the following:

Tier 1 Departures

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

This departure changes the STP plant-specific design by modifying Loop A of the RHR system to have a return to the fuel pool cooling system.

- STD DEP T1 2.12-2 — I&C Power Divisions

This departure adds a fourth division of safety-related power to the Class 1E instrument and control power supply system.

Tier 2 Departure Requiring Prior NRC Approval

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

This departure changes the STP design by incorporating two reserve auxiliary transformers (RATs) in place of one in the original ABWR design.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 1.1-2 — Dual Units at STP 3 & 4

This departure addresses the applicant's statement that the shared systems between STP Units 3 and 4 do not change the assessed risk associated with shutdown conditions.

- STP DEP 5.4-1 — Reactor Water Cleanup System

This departure addresses the STP Units 3 and 4 plant-specific design that requires a single CUW pump to provide 100 percent capacity during operating Modes 4 and 5. This is a change from the ABWR DCD design that requires both pumps.

- STP DEP 6C-1 — Containment Debris Protection for ECCS Strainers

The model of strainer changed from conical suction strainer to CCI cassette type strainer which satisfies the guidance in RG 1.82, Rev.3.

- STD DEP 10.4-5 Condensate and Feedwater System (Table 19L-9)

This departure changes the condensate and FW system by modifying the success criteria to include the condensate booster pumps.

19L.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19L.4 Technical Evaluation

NRC staff reviewed Appendix 19L of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "ABWR Shutdown Risk Evaluation."

The staff reviewed the information in the COL FSAR:

Tier 1 Departures

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

This departure changes the STP plant-specific design by modifying Loop A of the RHR system to have a return to the fuel pool cooling system. The staff agreed that increasing the number of RHR loops that connect to the fuel pool cooling and cleanup system (FPCCS) from two to three decreases the risk of a shutdown.

- STD DEP T1 2.12-2 I&C Power Divisions

This departure adds a fourth division of safety-related power to the Class 1E instrument and control power supply system. The staff agrees that this change represents an improvement and does not result in an increase in the risk of a shutdown.

Tier 2 Departures Requiring Prior NRC Approval

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

The STP design incorporates two RATs in place of one in the original ABWR design. The staff agrees that two RATs afford greater reliability for offsite AC power and therefore, decrease the frequency of a LOOP event.

The applicant states that these departures either improve the design and therefore decrease the CDF relative to the referenced ABWR design, or do not affect the CDF.

Tier 2 Departures Not Requiring Prior NRC Approval

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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 10.4-5 Condensate and Feedwater System (Table 19L-9)

Due to the change of the condensate and FW system, the success criteria are modified to include the condensate booster pumps. This change is reflected in Table 19L-9, "Dependency of Core Cooling Systems on Electrical Power." The change is acceptable.

- STD DEP 1.1-2 Dual Units at STP 3 & 4

The applicant states that the shared systems between STP Units 3 and 4 do not result in any changes to the assessed risk associated with shutdown conditions. The staff has questions on this statement. See the evaluation under *19.L.8, Loss of Decay Heat Removal Events* below.

- STP DEP 5.4-1 Reactor Water Cleanup System

In the STP plant-specific design, a single CUW pump is needed to provide 100 percent capacity during operating modes 4 and 5. This is a change from the original ABWR design, which requires both pumps. The change has no quantifiable effect on PRA. The staff agreed with this assessment.

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

The model of strainer changed from conical suction strainer to CCI cassette type strainer, which satisfies the requirements of RG 1.82, Rev.3. This departure addresses the applicant's statement that the ECCS suction strainer departure meets NRC requirements and does not increase the shutdown risk profile. The staff agreed with this assessment.

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

19L.6.4 Reactor Water Cleanup System

- STP DEP 5.4-1 Reactor Water Cleanup System

The staff evaluated this design change and agreed that it represents an improvement in the reliability of the CUW system and a reduction in the risk of a shutdown. The CUW can mitigate a loss of decay heat removal (DHR) after 8 days post-shutdown. The staff agreed with this assessment.

19L.6.5 Residual Heat Removal System

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

The ABWR RHR system is a closed system consisting of three independent pump loops that inject water into the vessel and/or remove heat from the reactor core or the containment. Loop A differs from Loops B and C in that the Loop A return line goes to the RPV through the FW line, whereas the return lines for Loops B and C go directly to the RPV. In this design change, all three RHR loops are connected to the fuel pool cooling and cleanup system instead of two loops for the referenced ABWR DCD, with normally close inter-ties to permit additional supplemental cooling during refueling outages. The staff agreed that increasing the number of RHR loops that connect to the FPCCS from two to three decreases the risk of a shutdown.

19L.6.6 Summary of Reactor Pressure Vessel Draining Events

- STP DEP 5.4-1 Reactor Water Cleanup System

See the discussion in Section 19.L.6.4.

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

See the discussion in Section 19L.6.5.

- STD DEP T1 2.12-2 I&C Power Divisions

The Instrument and Control Power Supply System described in the DCD Tier 1 provides power to three mechanical safety-related divisions (I, II, and III) and not to safety-related Distributed and Control and Information System (DCIS) Division IV. This departure adds a fourth division of safety-related power to the Class 1E instrument and control power supply system.

This design change represents an improvement and does not result in an increase in the risk of a shutdown. The staff agreed with this assessment.

19L.7.2 Success Criteria

- STD DEP 10.4-5 Condensate and Feedwater System

Not directly related to departure STD DEP 10.4-5 but referenced in Table 19L-9 of the STP Units 3 and 4 FSAR is a list of core cooling systems that satisfy the core cooling system success criteria. However, the Table 19L-9 list only contains pumps with the capability to keep the core covered. The core heat removal path is not listed, such as (1) the number of SRVs that need to be opened to remove heat from the vessel, or (2) where the core heat is to be discharged (e.g., the suppression pool) given an extended loss of DHR. The success criteria need to be augmented to include all SSCs in the heat removal path, not just the list of injection paths. The applicant's response to RAI 19-17 dated July 13, 2009 (ML092740559) states that the SSCs necessary for decay heat removal are included in DCD Section 19Q.7 and in Table 19Q-2 of the FSAR. The staff found this response acceptable. Therefore, RAI 19-17 is resolved.

19L.8 Loss of Decay Heat Removal Events

- STP DEP 5.4-1 Reactor Water Cleanup System

See the discussion in Section 19.L.6.4.

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

See the discussion in Section 19L.6.5.

- STD DEP 1.1-2 Dual Units at STP Units 3 & 4

The applicant stated that the shared systems between STP Units 3 and 4 do not result in any changes to the assessed risk associated with shutdown conditions. In the FSAR, the applicant stated that the shared fire water system between the STP Units 3 and 4 is not expected to result in any changes to the assessed risk associated with a shutdown, because the frequency for both units being in a shutdown condition and requiring backup cooling is extremely small.

However, there are currently no administrative controls precluding both units entering into a refueling outage or entering a forced shutdown simultaneously. In addition, the Abnormal Procedures for STP Units 1 and 2 require a plant shutdown before the arrival of a hurricane. NRC staff identified the need for additional information before concluding that the shared fire water system does not change the risk of a shutdown. The staff then issued RAI 19-18 requesting the applicant to evaluate quantitatively the CDF resulting from a postulated dual unit station blackout event, given a grid-related or severe weather LOOP (including hurricanes and tornadoes) during Modes 4 and 5.

The staff evaluated the applicant's response to RAI 19-18 dated July 13, 2009 (ML092740559) and found a screening evaluation that used a LOOP frequency of 0.1 per year. But this screening evaluation did not include equipment failures following a postulated hurricane event. The staff then issued RAI 19.01-31 requesting the applicant to provide the shutdown and the full-power hurricane CDF and a large early release frequency (LERF) [intended to be LRF] that considered the shared fire water system. The staff also requested a description of the dominant sequences contributing to the shutdown and the full-power hurricane CDF and LRF estimates. RAI 19.01-31 also included the unresolved issues from RAI 19-18. Therefore, RAI 19-18 is considered resolved and closed. RAI 19.01-31 was tracked as Open Item 19-9 in the SER with open items.

The applicant submitted the final response to RAI 19.01-31 (Open Item 19-9) on February 16, 2011 (ML110490542). In this response, the applicant listed hurricane mitigation strategies and documented the design-basis wind speed for the site as 134 mph for a 3-second gust. This design-basis wind speed is applied to the combustion gas turbine structure, the 345-KV switchyard, and the fire water pump house. The return period of the 3-second gust wind is one in a hundred years based on the American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) Standard ASCE/SEI 7–02, "Minimum Design Loads for Building and Other Structures." This response also included a simplified quantitative assessment to satisfy the requirements of 10 CFR 52.79(d)(1) to evaluate the effect of a hurricane on STP Units 3 &4 at the design basis wind speed. Crediting the compensatory measures documented in FSAR Section 19.4.6, ABWR Shutdown and COM 19.4-1 CR 10-15528 Action 2, the core damage frequency with credit for the ACIWA function and with only limited credit for the CTGs (failure likelihood of 0.5) was estimated to be 1.5E-8 per year. In this simplified screening assessment, no credit was given diesel generator recovery and use of RCIC. The estimated LRF would be less than or equal to this estimated CDF. Thus, the Commission's LRF goals of 1E-6/year for new reactors have been met with margin.

An additional sensitivity analysis was performed to evaluate the effects of hurricane winds that exceed the STP design basis windspeed. The wind speed recurrence interval selected was 200 years, which is approximately 142 mph using the methodology described in ASC/SEI 7-05. In this sensitivity analysis, the compensatory measures documented in FSAR Section 19.4.6, ABWR Shutdown and COM 19.4-1 CR 10-15528 Action 2, were credited. However, the ACIWA function and the CTGs were assumed to fail at this wind speed. Diesel generator recovery was credited based on operation of the RCIC system for at least 8 hours. The core damage frequency per unit was estimated to be 4.6E-7 per year.

This sensitivity study did not credit use of a portable diesel driven fire pump. This portable pump will be staged in an onsite Seismic Category I structure prior to arrival onsite of sustained winds in excess of 73 mph. This fire pump is described in Tier 2, Subsection 5.4.7 of the DCD. The portable pump and the valves that align this pump to the RHR system: F103C, F102C, and F101C are included in the reliability assurance program described in Section 19.K.11.5 of the

DCD and are included in Table 19K-4 of the DCD. This information was included by reference in the STP Units 3 and 4 FSAR. The fire pump can take suction from any available water source including the Fire Water Storage Tank and the Ultimate Heat Sink system. Crediting the use of this fire pump further reduces the estimated core damage frequency to less than 1E-8 per year. The estimated LRF would be less than or equal to this estimated CDF. Tus, the Commission's LRF goals of 1E-6/year for new reactors have been met with margin.

For both assessments, one EDG for each unit is assumed to be running and loaded on its Class 1E bus at least two hours before the onsite arrival of winds in excess of 73 mph. Therefore, the common cause failure parameters were adjusted to remove one EDG train from the start (diesel and ventilation fan), to run the first hour (diesel), and to close the output breakers for each unit. In response to RAI 19.01-31, this specific compensatory action and others credited in the risk evaluations are documented in FSAR Section 19.4.6, "ABWR Shutdown Risk," and COM 19.4-1 CR 10-15528 Action 2. These specific actions include developing an abnormal operating procedure for severe weather that is consistent with NUMARC 87-00, Revision 1, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Initiative 2, "Procedures," and Section 2.11, "Hurricane Preparations." This abnormal operating procedure will contain specific requirements as listed below:

- Action shall be initiated to place the units in Mode 3 (Hot Shutdown) at least 2 hours before wind speeds exceed 73 mph (or 96 mph as determined by discussions with the Transmission Distribution Service Provider (TDSP)). The applicability of this requirement is for units in Modes 1 and 2. Units in Modes 3, 4, or 5 will not be changed/maintained in Modes 3, 4, or 5.

- One EDG in each unit is started and loaded onto its safety bus, and the bus is disconnected from offsite power at least 2 hours before the onsite arrival of winds in excess of 73 mph.

- If an unstable electrical grid develops or is predicted by the TDSP, the remaining diesel generators are started and loaded on their safety buses, and the buses are disconnected from offsite power.

- If applicable for the current unit mode, the RCIC will be verified as available to provide core cooling in the event of a station blackout.

- The portable diesel driven fire pump will be staged in an onsite Seismic Category I structure prior to arrival onsite of sustained winds in excess of 73 mph.

- If the containment is inerted at the time of the hurricane warning, it will remain inerted during a forced shutdown due to a hurricane and in anticipation of restoring the units to operation after the hurricane has passed.

Based on (a) results of the quantitative assessment and sensitivity analysis that satisfy the requirements of 10 CFR 52.79(d)(1); (b) COM 19.4-1 to develop abnormal operating procedures for severe weather that will contain the specific requirements documented in FSAR Section 19.4.6; and (c) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-9 (RAI 19.01-31) are resolved. Verification that the applicant's proposed changes are in the revised FSAR sections is being tracked as **Confirmatory Item 19-15**.

- STD DEP T1 2.12-2 I&C Power Divisions

See the discussion in Section 19L.6.6.

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

The STP design incorporates two RATs in place of one in the original ABWR design.

The STP FSAR states that two RATs afford greater reliability for offsite AC power and therefore, decrease the frequency of a LOOP event. NRC staff agreed with the applicant.

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

The applicant states that the ECCS suction strainer departure meets NRC requirements and represents an improvement in the design.

NRC staff agreed that the improvement in the ECCS suction strainer design (1) addresses the staff's concerns noted in NRC Bulletins 93-02, GL 97-04, and GL 98-04; (2) is designed to meet the guidance referenced in RG 1.82, NUREG/CR–6224, NUREG/CR–6808, and Utility Resolution Guidance, NEDO 32686; and (3) is acceptable to the staff because this design decreases the risk of a shutdown.

19L.9.4 Loss of Fuel Pooling Cooling

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

See the discussion in Section 19L.6.5.

19L.5 Post Combined License Activities

There are no post COL activities related to this section.

19L.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "ABWR Shutdown Risk Evaluation." Pursuant to 10 CFR 52.63(a)(5) and Part 52 Appendix A Section VI.B.1, all nuclear safety issues relating to "ABWR Shutdown Risk Evaluation" that were incorporated by reference have been resolved. With the exception of **Confirmatory Item 19-15**, no outstanding information is expected to be addressed in the COL FSAR related to this section. In addition, based on the above discussion on "ABWR Shutdown Risk Evaluation," the staff concluded, pending the completion of Confirmatory Item 19-15, that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19M Fire Protection Probabilistic Risk Assessment (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.5.2.1, "Description of the Internal Fire Risk Evaluation"; 19.1.5.2.2, "Results from the Internal Fire Risk Evaluation"; 19.2.2, "Severe Accident Prevention.")

19M.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19M of the ABWR DCD due to the departures of the STP Units 3 and 4 design from those described in the ABWR DCD.

19M.2 Summary of Application

Appendix 19M of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19M of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19M, the applicant provides the following:

Tier 1 Departures

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure addresses the applicant's statement that changes to the RCIC pump reduce the overall risk of fire. The new RCIC pump design is expected to increase RCIC reliability and reduce overall risk. This reduction also occurs in the results assessing the risk of fire, due to the importance of the RCIC pump operation following a control room fire.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 1.1-2 Dual Units at STP Units 3 & 4

This departure addresses the statement that the use of a shared fire protection pump house and storage tanks does not affect the FIVE analysis results.

- STD DEP 1.2-1 Control Building Annex

This departure moves the reactor internal pump motor generator (MG) sets and their switchgear from the control building to the control building annex. The applicant states that the relocation of MG sets lowers the ignition frequencies for the fire compartment in the control building.

- STD DEP 1.2-2 Turbine Building

This departure addresses the applicant's statement that the turbine building modifications will not affect the generic fire frequencies used to perform the FIVE analyses described in the various FSAR Chapter 19 sections. Furthermore, changes to turbine building design will not affect the LOOP event models used to quantify the effects of fire in the turbine building.

19M.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19M.4 Technical Evaluation

NRC staff reviewed Appendix 19M of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL 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 "Fire Protection Probabilistic Risk Assessment."

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

- STD DEP T1 2.4-3 RCIC Turbine/Pump

The applicant states that changes to the RCIC pump reduce the overall risk of fire. The new RCIC pump design is expected to increase RCIC reliability and reduce overall risk. This reduction also occurs in the results assessing the risk of fire due to the importance of the RCIC pump operation following a control room fire. The staff agreed with this statement.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 1.1-2 Dual Units at STP Units 3 & 4

The applicant states that the use of a shared fire protection pump house and storage tanks does not affect the FIVE analysis results. The applicant's evaluation of this departure described above, in accordance with Item B.5 of Section VIII, determined that this departure does not require prior NRC approval. The staff reviewed the Departures Report regarding this departure, and could not determine whether it is reasonable for this departure not to require prior NRC approval. Therefore, NRC staff issued RAI 19-7 requesting the applicant to clarify that human action is required for a manual switchover and to describe the impact on the risk of fire.

The applicant's supplemental response to RAI 19-7 dated December 3, 2009 (ML093421266) indicates that Table 19.2-2 of STP FSAR Tier 2 will be revised to state that there is no significant effect on CDF, no change to the PRA, and editorial changes to the fire protection system. NRC staff found this RAI response sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. The staff confirmed that the proposed revision is incorporated into Chapter 19 of FSAR Revision 4. Therefore, RAI 19-7 is resolved.

- STD DEP 1.2-1 Control Building Annex

This departure moves the reactor internal pump MG sets and their switchgear from the control building to the control building annex. The applicant states that the relocation of the MG sets lowers the ignition frequencies for the fire compartment in the control building. The applicant's evaluation of this departure described above, in accordance with Item B.5 of Section VIII, determined that this departure does not require prior NRC approval. The staff reviewed the Departures Report regarding this departure, and could not determine whether it is reasonable for this departure not to require prior NRC approval. Therefore, NRC staff issued RAI 19.01-16 asking the applicant to clarify that this new building is included in an evaluation of the risk of fire. In the response to this RAI dated August 5, 2009 (ML092220163), the applicant states that 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.

new control building annex is not safety related and does not include any safety-related equipment. For this reason, this building is not included in the internal fire analysis. The staff found this response acceptable, and RAI 19.01-16 is resolved.

- STD DEP 1.2-2 Turbine Building

The applicant states that the potential turbine building modifications do not affect the generic fire frequencies used to perform the FIVE analyses described in the various FSAR Chapter 19 sections. Also, potential changes to the turbine building design do not affect the LOOP event models used to quantify the effects of fire in the turbine building.

The applicant's evaluation of this departure described above, in accordance with Item B.5 of Section VIII, determined that this departure does not require prior NRC approval. The staff reviewed the Departures Report regarding this departure, and noticed that there are additional components with new locations in the STP turbine building. Therefore, the staff issued RAI 19.01-23 requesting the applicant to detail the risk of fire as a result of these changes. This RAI was tracked as Open Item 19-10 in the SER with open items. In the response to RAI 19.01-23 dated March 16, 2010 (ML100770391), the applicant states that no new equipment affects the safe shutdown for the STP design using the FIVE methodology. The equipment associated with the CTG is identified as new equipment in the STP design. However, the CTG was screened out from the analysis in the SSAR because fires in those areas do not directly lead to a plant trip and do not affect offsite power distribution to the plant using the FIVE methodology. Therefore, the new equipment and the new location of the equipment in the turbine building do not affect the ABWR DCD conclusion using the FIVE methodology. The staff found this response acceptable, and this RAI is resolved.

In RAI 19-19, the staff also asked the applicant to explain whether the analysis assessing the risk of fire includes the RSW pump house. In the response to this RAI dated August 18, 2009 (ML092310681), the applicant states that the RSW pump house is part of the intake structure and is evaluated in the FIVE analysis in the ABWR DCD. The requirement for the intake structure is also documented in DCD Section 9.5.1. Tier 1, Chapter 2.11.9 lists the RSW system interface requirements; and Item (2) describes the fire barrier requirements, which include interdivisional boundaries (e.g., walls, floors, doors, and penetrations) that have a 3-hour fire rating. These requirements are unchanged in the STP Units 3 and 4 COL FSAR. The staff found this response acceptable, and RAI 19-19 is resolved.

19M.5 Post Combined License Activities

The applicant identifies commitment (COM 19.9-11) to address COL License Information Item 19.12 as discussed in SER Section 19.9.4.

19M.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Fire Protection Probabilistic Risk Assessment." 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 "Fire Protection Probabilistic Risk Assessment" that were incorporated by reference have been resolved. In addition, based on the above discussion on "Fire Protection Probabilistic Risk Assessment," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19N <u>Analysis of Common-Cause Failure of Essential Communications Equipment</u>

19N.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19N of the U. S. ABWR DCD due to the departures of the STP Units 3 and 4 design from those described in the ABWR DCD.

19N.2 Summary of Application

Appendix 19N of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19N of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19N, the applicant provides the following:

<u>Tier 1 Departure</u>

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

This departure introduces a new safety-related I&C architecture that causes the elimination of the obsolete data communication technology; elimination of unnecessary inadvertent actuation prevention logic and equipment; clarification of digital control systems and nomenclature; and implementation architectural changes and testing and surveillance changes to the SSLC. Chapter 7 of this SER provides additional information on this departure.

<u>Supplemental Information</u>

In Section 19N.1, “Introduction,” the applicant updates a supplemental study to further investigate the effects from the use of common instruments, ECF equipment, transmission networks for reactivity control (scram), ECCS (core cooling and decay heat removal), and LDIS (isolation).

In Section 19N.4, “Potential Causes and Defenses Against ECF CC,” the applicant incorporates this section by reference with the standard departure numbered STD DEP T1 3.4-1.

In the following sections and subsections, the applicant updates the nomenclature used in the text:

- 19N.2 Results and Conclusions
- 19N.3 Basis for the Analysis
- 19N.4.1 Earthquake
- 19N.4.2 Loss of D.C. Power
- 19N.4.3 Loss of Cooling
- 19N.4.4 Sensor Miscalibration
- 19N.4.5 Remote DLC Miscalibration
- 19N.4.7 Maintenance/Test Error
- 19N.4.9 Electromagnetic Interference (EMI)
- 19N.4.10 Fire
- 19N.4.11 Software
- 19N.4.12 Summary
- 19N.5 Discussion of the Effect on Core Damage Frequency

- 19N.5.1 General Plant Transient Events
- 19N.5.2 Loss of Feedwater Event
- 19N.5.3 Loss of Coolant Accidents
- 19N.5.4.1 Loss of Offsite Power
- 19N.5.4.2 Loss of DC Power
- 19N.5.4.3 Inadvertent Open Relief Valve
- 19N.5.4.4 Loss of Service Water
- 19N.5.4.5 Loss of Instrument Air
- 19N.5.5 CCF of ECF During Normal Plant Operation
- 19N.6 Discussion of the Effect on Isolation Capability
- 19N.7 Summary

19N.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in section 19.1.3 of this SER.

19N.4 Technical Evaluation

NRC staff reviewed Appendix 19N of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic[1]. The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Analysis of Common-Cause Failure of Essential Communications Equipment."

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

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

NRC staff reviewed STD DEP T1 3.4-1, which is included in Appendix 19N of the STP Units 3 and 4 COL FSAR. The staff determined that the specific text changes in Appendix 19N are appropriate and address Departure STD DEP T1 3.4-1. Within the review scope of this section, the staff found that this departure is acceptable and editorial in nature.

Supplemental Information

19N.1 Introduction

The applicant updates the nomenclature used in the text and SSLC descriptions.

19N.2 Results and Conclusions

The applicant updates the nomenclature used in the text.

[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.

19N.3 Basis for the Analysis

The applicant updates the nomenclature used in the text.

NRC staff issued RAI 19-23 requesting the applicant to address inconsistencies between the STP Units 3 and 4 Departures Report and the STP FSAR Revision 2, and to revise the STP Units 3 and 4 FSAR (as necessary).

The applicant's response to RAI 19-23 dated August 26, 2009 (ML092430135), states that Appendix 19N of the STP Units 3 and 4 COL FSAR Tier 2 will be revised to address the COL application changes stated in the RAI response. The staff found this response to RAI 19-23 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. The staff confirmed that the proposed revision is incorporated into Chapter 19 of FSAR Revision 4. Therefore, this issue in RAI 19-23 is resolved.

19N.4 Potential Causes and Defenses Against ECF CCF

The applicant incorporates this section by reference with the standard departure numbered STD DEP T1 3.4-1.

19N.4.1 Earthquake

The applicant updates the nomenclature used in the text.

19N.4.2 Loss of DC Power

The applicant updates the nomenclature used in the text.

19N.4.3 Loss of Cooling

The applicant updates the nomenclature used in the text.

19N.4.4 Sensor Miscalibration

The applicant updates the nomenclature used in the text.

19N.4.5 Remote DLC Miscalibration

The applicant updates the nomenclature used in the text.

NRC staff issued RAI 19-23 requesting the applicant to address inconsistencies between the STP Units 3 and 4 Departures Report and the STP Units 3 and 4 FSAR Revision 2, and to revise the STP Units 3 and 4 FSAR (as necessary).

The applicant's response to RAI 19-23 dated August 26, 2009 (ML092430135), indicates that Appendix 19N of the STP Units 3 and 4 COL Tier 2 FSAR will be revised to address the COL application changes stated in the RAI response. The staff found this response to RAI 19-23 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. The staff confirmed that the proposed revision is incorporated into Chapter 19 of FSAR Revision 4. Therefore, this issue in RAI 19-23 is resolved.

19N.4.7 Maintenance/Test Error

The applicant updates the nomenclature used in the text.

19N.4.9 Electromagnetic Interference (EMI)

The applicant updates the nomenclature used in the text.

19N.4.10 Fire

The applicant updates the nomenclature used in the text.

19N.4.11 Software

The applicant updates the nomenclature used in the text.

19N.4.12 Summary

The applicant updates the nomenclature used in the text.

19N.5 Discussion of the Effect on Core Damage Frequency

The applicant updates the nomenclature used in the text.

NRC staff issued RAI 19-23 requesting the applicant to address inconsistencies between the STP Units 3 and 4 Departures Report and the STP Units 3 and 4 FSAR Revision 2, and to revise the STP Units 3 and 4 FSAR (as necessary).

The applicant's response to RAI 19-23 dated August 26, 2009, indicates that Appendix 19N of the STP Units 3 and 4 COL FSAR Tier 2, will be revised to address the COL application changes stated in the RAI response. The staff found this response to RAI 19-23 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. The staff confirmed that the proposed revision is incorporated into Chapter 19 of FSAR Revision 4. Therefore, this issue in RAI 19-23 is resolved.

19N.5.1 General Plant Transient Events

The applicant updates the nomenclature used in the text.

NRC staff issued RAI 19-23 requesting the applicant to address inconsistencies between the STP Units 3 and 4 Departures Report and the STP Units 3 and 4 FSAR Revision 2, and to revise the STP Units 3 and 4 FSAR (as necessary).

The applicant's response to RAI 19-23 dated August 26, 2009, indicates that Appendix 19N of the STP Units 3 and 4 COL FSAR Tier 2, will be revised to address the COL application changes stated in the RAI response. The staff found this response to RAI 19-23 sufficient to meet the guidance in RG 1.206 and SRP Chapter 19. Verification that the proposed revision is incorporated into the next FSAR update was tracked as a confirmatory item in the SER with open items. The staff confirmed that the proposed revision was incorporated into Chapter 19 of FSAR Revision 4. Therefore, this issue in RAI 19-23 is resolved.

19N.5.2 Loss of Feedwater Event

The applicant updates the nomenclature used in the text.

19N.5.3 Loss of Coolant Accidents

The applicant updates the nomenclature used in the text.

19N.5.4 Other Initiating Events

The applicant makes no changes to this section.

19N.5.4.1 Loss of Offsite Power

The applicant updates the nomenclature used in the text.

19N.5.4.2 Loss of DC Power

The applicant updates the nomenclature used in the text.

19N.5.4.3 Inadvertent Open Relief Valve

The applicant updates the nomenclature used in the text.

19N.5.4.4 Loss of Service Water

The applicant updates the nomenclature used in the text.

19N.5.4.5 Loss of Instrument Air

The applicant updates the nomenclature used in the text.

19N.5.5 CCF of ECF During Normal Plant Operation

The applicant updates the nomenclature used in the text.

19N.6 Discussion of the Effect on Isolation Capability

The applicant updates the nomenclature used in the text.

19N.7 Summary

The applicant updates the nomenclature used in the text.

19N.5 Post Combined License Activities

The applicant identifies commitment (COM19.9-7) to address COL License Information Item 19.8 as discussed in SER Section 19.9.4.

19N.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to “Analysis of Common-Cause Failure of Essential Communications Equipment.” 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 “Analysis of Common-Cause Failure of Essential Communications Equipment” that were incorporated by reference have been resolved.

In addition, based on the above discussion on the “Analysis of Common-Cause Failure of Essential Communications Equipment,” the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19O Not Used

This appendix is not used in the ABWR DCD or in the applicant’s FSAR.

19P Evaluation of Potential Modifications to the ABWR Design

Appendix 19P of the STP COL FSAR incorporates by reference with no departures or supplements Appendix 19P, “Evaluation of Potential Modifications to the ABWR Design,” of Revision 4 of the ABWR DCD, which is incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this appendix remains for review.[1] The staff’s review confirmed that there is no outstanding issue related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the “Evaluation of Potential Modifications to the ABWR Design” have been resolved.

19Q ABWR Shutdown Risk Assessment (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.6.1, “Description of the Low-Power and Shutdown Operations PRA”; and 19.1.6.2, “Results from the Low-Power and Shutdown Operations PRA.”)

19Q.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19Q of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

19Q.2 Summary of Application

Appendix 19Q of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19Q of the ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19Q, 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 2.4-1 RHR System and Spent Fuel Pool Cooling

This departure changes the RHR design of the STP Units 3 and 4 design to three RHR loops connected to the FPCCS instead of the two RHR loops in the original ABWR design.

- STD DEP T1 2.12-2 I&C Power Divisions

This departure adds a fourth division of safety-related power to the Class 1E instrument and control power supply system.

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

This departure changes the safety-related I&C Architecture, such as eliminating obsolete data communication technology. This departure eliminates references to the EMS and the NEMS, and replaces them with separate and independent system level data communication capabilities. The departure also eliminates references to multiplexed functions of plant systems and the plant layout in relation to the risk of an ABWR fire.

- STP DEP T1 5.0-1 Site Parameters

This departure addresses the applicant's analysis of external flooding at STP Units 3 and 4 for power operation documented in Appendix 19R. The applicant states that the incremental increase in risk during a shutdown due to external flooding is very small because of the fraction of time the plant is in a shutdown condition during a year and the small likelihood of an external flood occurrence during shutdown conditions. The applicant states that the ABWR DCD remains bounding for the risk of a shutdown.

Tier 2 Departure Requiring Prior NRC Approval

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

This departure changes the STP design by incorporating two RATs in place of the one RAT in the ABWR original design.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 10.4-5 Condensate and Feedwater System

This departure addresses the applicant's statement that if all RHR systems failed, the RPV would pressurize and the main condenser could be made available by opening the MSIVs; drawing a vacuum in the condenser; and operating the feedwater, condensate booster, and condensate pumps for makeup.

- STD DEP 1.1-2 Dual Units at STP Units 3 & 4

This departure addresses the applicant's statements that the shared systems between STP Units 3 and 4 do not result in any changes to the assessed risk associated with shutdown conditions.

- STD DEP 5.4-1 Reactor Water Cleanup System

This departure addresses changes in the STP plant-specific design to the need for a single CUW pump to operate and provide 100 percent capacity during operating Modes 4 and 5. The original ABWR design requires both pumps.

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

This departure addresses the applicant's statement that the ECCS suction strainer departure meets NRC requirements and does not result in an increase in the shutdown risk profile.

19Q.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER

19Q.4 Technical Evaluation

NRC staff reviewed Appendix 19Q of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "ABWR Shutdown Risk Assessment."

The staff reviewed the information in the COL FSAR:

Tier 1Departures

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

This departure changes the RHR design of the STP Units 3 and 4 design to three RHR loops connected to the FPCCS instead of the two RHR loops in the original ABWR design. The staff agreed that increasing the number of RHR loops that connect to the FPCCS from two to three decreases the risk of a shutdown.

- STD DEP T1 2.12-2 I&C Power Divisions

This departure adds a fourth division of safety-related power to the Class 1E instrument and control power supply system. The staff agrees that this change represents an improvement and does not result in an increase in the risk of a shutdown.

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

This departure changes the safety-related I&C Architecture, such as eliminating obsolete data communication technology. This departure eliminates references to the EMS and the NEMS, and replaces them with separate and independent system level data communication capabilities. The departure also eliminates references to multiplexed functions of plant systems

[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.

and the plant layout in relation to the risk of an ABWR fire. The evaluation of departure has been performed in Section 19.1.4 of this SER.

- STP DEP T1 5.0-1 Site Parameters

This departure addresses the applicant's analysis of external flooding at STP Units 3 and 4 for power operation documented in Appendix 19R. The applicant states that the incremental increase in risk during a shutdown due to external flooding is very small because of the fraction of time the plant is in a shutdown condition during a year and the small likelihood of an external flood occurrence during shutdown conditions. The applicant states that the ABWR DCD remains bounding for the risk of a shutdown. See Section 19R.4 for the evaluation summary.

Tier 2 Departure requiring Prior NRC Approval

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

The STP design incorporates two RATs in place of one in the original ABWR design. The staff agrees that two RATs afford greater reliability for offsite AC power and therefore, decrease the frequency of a LOOP event.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 10.4-5 Condensate and Feedwater System

This departure addresses the applicant's statement that if "all RHR systems failed, the RPV would pressurize and the main condenser could be made available by opening the MSIVs; drawing a vacuum in the condenser; and operating the FW, condensate booster, and condensate pumps for makeup." The staff agreed with the changes of the text from the DCD.

- STD DEP 1.1-2 Dual Units at STP Units 3 & 4

This departure addresses the applicant's statements that the shared systems between STP Units 3 and 4 do not result in any changes to the assessed risk associated with shutdown conditions. The staff has questions on this statement. See Subsection 19Q.4.4 below for discussion.

- STD DEP 5.4-1 Reactor Water Cleanup System

This departure addresses changes in the STP plant-specific design to the need for a single CUW pump to operate and provide 100 percent capacity during operating Modes 4 and 5. The original ABWR design requires both pumps. The change has no quantifiable effect on PRA. The staff agreed with this assessment.

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

The model of strainer changed from conical suction strainer to CCI cassette type strainer which satisfies the requirements of RG 1.82, Rev.3. This departure addresses the applicant's statement that the ECCS suction strainer departure meets NRC requirements and does not increase the shutdown risk profile. The staff agreed with this assessment.

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

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

19Q.3 Summary of Results

• STD DEP T1 2.4-1	Residual Heat Removal System and Spent Fuel Pool Cooling
• STD DEP T1 2.12-2	I&C Power Divisions
• STP DEP T1 3.4-1	Safety-Related I&C Architecture
• STP DEP T1 5.0-1	Site Parameters
• STD DEP 1.1-2	Type of License Required
• STD DEP 8.3-1	Plant Medium Voltage Electrical System Design
• STD DEP 5.4-1	Reactor Water Cleanup System
• STP DEP 6C-1	Containment Debris Protection for ECCS Strainers

The applicant states that these departures either (1) improve the design and therefore decrease the CDF relative to the referenced ABWR design, or (2) do not affect the CDF. The staff agrees with this assessment.

19Q.4.1 Decay Heat Removal

The applicant provides other potential heat sinks, including the suppression pool, RWCS, or the FPCCS.

- STD DEP 5.4-1 — Reactor Water Cleanup System

In the STP plant-specific design, a single CUW pump is needed to operate and provide 100-percent capacity during operating Modes 4 and 5, which is a change from the original ABWR design that requires both pumps. NRC staff evaluated this design change and agreed that it represents an improvement in the reliability of the CUW system and a reduction in the risk of a shutdown. The CUW can mitigate a loss of DHR after 8 days post-shutdown. The change has no quantifiable effect on PRA. The staff agrees with this assessment.

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

In the STP plant-specific design, Loop A of the RHR system is modified to have a return to the fuel pool cooling system. In this design change, all three RHR loops are connected to the fuel pool cooling and cleanup system instead of the two loops in the referenced ABWR DCD, with normally close inter-ties to permit additional supplemental cooling during refueling outages. NRC staff agreed that additional supplemental cooling to the fuel pool decreases the risk of a shutdown.

19Q.4.2 Inventory Control

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

In the STP plant-specific design, Loop A of the RHR system is modified to have a return to the fuel pool cooling system. In this design change, all three RHR loops are connected to the fuel pool cooling and cleanup system instead of the two loops in the referenced ABWR DCD, with normally close inter-ties to permit additional supplemental cooling during refueling outages to reduce outage time. NRC staff agreed that additional supplemental cooling to the fuel pool decreases the risk of a shutdown.

- STP DEP 6C-1 — Containment Debris Protection for ECCS Strainers

The applicant states that the ECCS suction strainer departure meets NRC requirements and does not result in an increase in the shutdown risk profile. NRC staff agrees that an improvement in the ECCS suction strainer design (1) addresses the staff's concerns noted in NRC Bulletins 93-02, GL 97-04, and GL 98-04; (2) is designed to the guidance referenced in RG 1.82, NUREG/CR 6224, NUREG/CR 6808, and Utility Resolution Guidance, NEDO 32686; and (3) is acceptable to the staff because there would be a decrease in the risk of a shutdown.

19Q.4.4 Electrical Power

The applicant states that in the event that one phase of the main transformer were to fail, an installed spare is available to return the preferred source of offsite power to service without any delays.

- STD DEP 1.1-2 — Dual Units at STP Units 3 & 4

The applicant stated that the shared systems between STP Units 3 and 4 does not result in any changes to the assessed risk associated with shutdown conditions, because the frequency that both units will be in a shutdown condition and will require backup cooling is extremely small. However, there are currently no administrative controls preventing both units from entering into a refueling outage or entering a forced shutdown simultaneously. Also, the Abnormal Procedures for STP Units 1 and 2 require a plant shutdown before the arrival of a hurricane. Therefore, NRC staff needed additional information before concluding that the shared fire water system does not result in any change to the risk of a shutdown. The staff issued RAI 19-18 requesting the applicant to evaluate quantitatively the core damage frequency resulting from a postulated dual unit station blackout event, given a grid-related or severe weather LOOP (including hurricanes and tornadoes) during operating Modes 4 and 5.

The staff evaluated the applicant's response to RAI 19-18 and found a screening evaluation that used a LOOP frequency of 0.1 per year. But this screening evaluation did not include equipment failures following a postulated hurricane event. The staff then issued RAI 19.01-31 requesting the applicant to provide the shutdown and the full-power hurricane CDF and LERF [intended to be LRF] that considered the shared fire water system. The staff also requested a description of the dominant sequences contributing to the shutdown and the full-power hurricane CDF and LRF estimates. The staff issued RAI 19.01-31 to include the unresolved issues of RAI 19-18. Therefore, RAI 19-18 is considered resolved and closed. RAI 19.01-31 was tracked as Open Item 19-9 in the SER with open items.

The applicant submitted the final response to RAI 19.01-31 (Open Item 19-9) on February 16, 2011 (ML110490542). In this response, the applicant listed the hurricane mitigation strategies and documented the design wind speed for the site, 134 mph for a 3-second gust. This design-basis wind speed is applied to the combustion gas turbine structure, the 345-KV switchyard, and the fire pump house. The return period of the 3-second gust wind is one in one hundred years based on ASCE/SEI-7–02. The staff's review of this RAI response is discussed in detail under "Hurricane Risk" later in this section. In summary, based on (a) results of the quantitative assessment and sensitivity analysis that satisfy the requirements of 10 CFR 52.79(d)(1); (b) COM 19.4-1 to develop abnormal operating procedures for hurricanes that will contain the specific requirements documented in FSAR Section 19.4.6; and (c) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-9 (RAI 19.01-31) are resolved. Verification that the changes are in the revised FSAR sections is being tracked as **Confirmatory Item 19-15**.

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

The STP Units 3 and 4 design incorporates two RATs in place of the one RAT in the ABWR original design. The STP Units 3 and 4 FSAR states that two RATs afford greater reliability for offsite AC power and therefore, a decrease in the frequency of a LOOP event. NRC staff agreed.

- STD DEP T1 2.12-2 I&C Power Divisions

This design departure adds a fourth division of safety-related power to the Class IE instrument and control power supply system. NRC staff agreed that increasing the number of safety-related divisions from three to four improves reliability and decreases the risk of a shutdown.

19Q.6 Flooding and Fire Protection

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

This departure changes safety-related I&C Architecture, including the elimination of obsolete data communication technology. This departure eliminates references to the EMS and the NEMS, which are replaced with separate and independent system level data communication capabilities. The departure also eliminates references to multiplexed functions of plant systems and plant layout in relation to the risk of an ABWR fire. The evaluation of departure has been performed in Section 19.1.4 of this SER.

Internal Floods

The applicant states that the fire barriers will prevent water due to flooding from non-divisional sources from entering a division area and will contain water in the fire area from divisional sources. The applicant also states that the practice of not routing unlimited sources of water (e.g., service water) through ECCS room areas and ensuring that other large water sources (e.g., suppression pool) can be contained will be beneficial in the event of a flood.

The applicant also reviews all ABWR sources of an internal flood and concludes that during shutdown conditions, at least one safety division will be unaffected by water damage from any postulated flood. Besides separation, features that contribute to these results include adequately sized room floor drains, water level alarms and the automatic isolation of flood sources for potentially affected rooms, mounted motors and other electrical equipment at least 20.32 cm above floor level, and water-tight doors. Administrative controls will be implemented to assure that at least one safety division with intact barriers is available at all times during a plant shutdown. For RSW pump house floods, the water-tight doors for the pump rooms and electrical equipment rooms are capable of withstanding floods from either direction.

External Floods

- STP DEP T1 5.0-1 Site Parameters

Appendix 19R presents the analysis performed for external flooding at STP Units 3 and 4 for power operation. The analysis considered the cascading failure of the upstream dams on the Colorado River, probable maximum precipitation (PMP) events, main cooling reservoir breach, and tsunamis. The breach of the main cooling reservoir is the design-basis flood for STP Units 3 and 4. If external flood barriers are open or removed and cannot be restored before high water levels reach the site, then core damage is assumed. An operating procedure for severe external flooding will be developed and implemented before fuel loading (COM 19.9-3). The applicants states that an incremental increase in risk during a shutdown from external flooding is very small because of the fraction of time the plant is in a shutdown condition during a year and the small likelihood of an external flood occurrence during shutdown conditions. The applicant states that the ABWR DCD remains bounding for the risk of a shutdown.

Although site-specific internal and external full power flooding sequences are evaluated in Appendix 19.R of the STP FSAR, there is no risk analysis or estimation of the site-specific shutdown frequency of internal and external floods. NRC staff then issued RAI 19-21 requesting the applicant to provide a quantitative site-specific shutdown risk assessment from internal and external floods that determines the CDF and LRF.

The applicant's response to RAI 19-21 dated August 18 2009 (ML092310681), states that internal flooding during a shutdown is evaluated in Section 19Q.6 of the DCD. Procedural controls were identified as a necessary barrier to prevent and minimize the effects of flooding, and are incorporated by reference in FSAR Appendix 19Q. The applicant also notes that the external flooding analysis described in Appendix 19R of the STP Units 3 and 4 COL FSAR was performed on an annual frequency and, therefore, was performed independent of the operating mode. The applicant references the results of the external flooding assessment for a postulated breach of the main cooling reservoir in response to RAI 19.01-10.

The applicant's response to RAI 19.01-10 dated July 23, 2009 (ML092080083), stated that additional design requirements identified for the RSW pump rooms in FSAR Section 19Q.6 ensure that the DCD internal flood assessment for shutdown conditions (including procedural controls) remains bounding for STP Units 3 and 4. The applicant also provided the results of the external flood assessment for the main cooling reservoir breach design-basis flooding event.

NRC staff found the applicant's response acceptable regarding the risk of a shutdown from internal flooding. However, based on staff's review of the results of the external flooding assessment and the detailed screening evaluation for breaching the main cooling reservoir, the staff issued RAI 19-30 requesting additional information on the probabilities used for this evaluation. This RAI was tracked as Open Item 19-12 in the SER with open items, and is discussed in detail in Section 19R.4 of this SER.

In response to Open Item 19-12, the applicant submitted the final response to RAI 19-30 on July 28, 2010 (ML102110184). This RAI response changes the status for all watertight doors and hatches to be normally closed. This RAI response also uses ASME/ANS RA-Sa-2009, Section 6-2.3, "The Fundamental Criteria for Screening External Events Other Than Fire and Seismic Events." Criterion (a) was used to screen external flood scenarios from a detailed quantitative evaluation. Criterion (a) is satisfied since the STP design for safety-related systems, structures, and components satisfies the requirements of SRP Section 3.4.2, Revision 3, which was in effect at the time of the COL application. Based on (1) the change in the watertight door status to be normally closed, and (2) the removal of the screening quantification for a postulated main cooling reservoir breach, and (3) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-12 (RAI 19-30) have been resolved. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR. Therefore, the staff found the applicant's modeling of external floods acceptable.

Hurricane Risk

NRC staff reviewed the risk of a shutdown from a postulated hurricane outlined in Section 19.Q.6 of the STP Units 3 and 4 FSAR Revision 2. The Abnormal Procedure for STP Units 1 and 2, which covers hurricanes, requires a plant shutdown before the onsite arrival of hurricane winds in excess of 73 miles per hour. In order to reduce the risk when responding to an approaching hurricane, the applicant commits to develop before fuel loading a procedure to cope with impending hurricanes (COM 19Q-1). The applicant states that the tornado analysis in the referenced ABWR DCD will bound the hurricane analysis with respect to high winds. The staff then noted that there is no site-specific analysis to support this assumption. The staff issued RAI 19-20 requesting the applicant to provide a quantitative site-specific, high winds shutdown risk assessment that determines the high winds-induced CDF and LERF [intended to be LRF] given the shared fire water system. The applicant's response to RAI 19-20 dated August 18, 2009 (ML092310681), clarified that since the STP Units 3 and 4 site are within the site parameters defined in the DCD and the high winds that were reviewed as part of the DCD approval, the paragraphs addressing "Hurricane Risk" were removed from FSAR Section 19Q.6, Revision 3. Also, the associated FSAR commitment (COM19Q-1) was deleted.

The staff evaluated the applicant's responses to RAI 19-18 (that was discussed in previous sections) and RAI 19-20 and concluded that the shared fire water system design departure (STD DEP 1.1-2) impacts the shutdown and the full-power hurricane risk assessment for the site. The staff then issued RAI 19.01-31 requesting the applicant to provide the following information in accordance with 10CFR Part 52.79(d)(1): (1) The shutdown and full-power hurricane CDF and LERF [intended to be LRF] estimates, (2) A description of the dominant sequences contributing to the shutdown and full-power hurricane CDF and LERF estimates, (3) The list of SSCs that are identified as risk significant for the RAP with the supporting FV and RAW for component basic events, human error probabilities, and CCFs.

The staff issued RAI 19.01-31 to include the unresolved issues of RAIs 19-18 and 19-20. Therefore, RAIs 19-18 and 19-20 are considered resolved and closed. RAI 19.01-31 was tracked as Open Item 19-9 in the SER with open items. The applicant submitted their final response to RAI 19.01-31 (Open Item 19-9) on February 16, 2011 (ML110490542). In this response, the applicant listed hurricane mitigation strategies and documented the design-basis wind speed for the site as 134 mph for a 3-second gust. This design-basis wind speed is applied to the combustion gas turbine structure, the 345-KV switchyard, and the fire water pump house. The return period of the 3-second gust wind is one in a hundred years based on the American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) Standard

ASCE/SEI 7–02, “Minimum Design Loads for Building and Other Structures.” This response also included a simplified quantitative assessment to satisfy the requirements of 10 CFR 52.79(d)(1) to evaluate the effect of a hurricane on STP Units 3 &4 at the design basis wind speed. Crediting the compensatory measures documented in FSAR Section 19.4.6, “ABWR Shutdown Risk,” and COM 19.4-1 CR 10-15528 Action 2, the core damage frequency with credit for the ACIWA function and with only limited credit for the CTGs (failure likelihood of 0.5) was estimated to be 1.5E-8 per year. In this simplified, screening, assessment, no credit was given diesel generator recovery and use of RCIC. The estimated LRF would be less than or equal to this estimated CDF. Thus, the Commission’s goals of 1E-6/year for new reactors have been met with margin.

An additional sensitivity analysis was performed to evaluate the effects of hurricane winds that exceed the STP design basis windspeed. The wind speed recurrence interval selected was 200 years, which is approximately 142 mph using the methodology described in ASC/SEI 7-05. In this sensitivity analysis, the compensatory measures documented in FSAR Section 19.4.6, ABWR Shutdown and COM 19.4-1 CR 10-15528 Action 2 were credited. However, the ACIWA function and the CTGs were assumed to fail at this wind speed. Diesel generator recovery was credited based on operation of the RCIC system for at least 8 hours. The core damage frequency per unit was estimated to be 4.6E-7 per year.

This sensitivity study did not credit use of a portable diesel driven fire pump. This portable pump will be staged in an onsite Seismic Category I structure prior to arrival onsite of sustained winds in excess of 73 mph. This fire pump is described in Tier 2, subsection 5.4.7 of the DCD. The portable pump and the valves that align this pump to the RHR system: F103C, F102C, and F101C are included in the reliability assurance program described in Section 19.K.11.5 of the DCD and are included in Table 19K-4 of the DCD. This information was included by reference in the STP Units 3 and 4 FSAR. The fire pump can take suction from any available water source including the Fire Water Storage Tank and the Ultimate Heat Sink system. Crediting the use of this fire pump further reduces the core damage frequency to less than 1E-8 per year. The estimated LRF would be less than or equal to this estimated CDF. Thus, the Commission’s goals of 1E-6/year for new reactors have been met with margin.

For both assessments, one EDG for each unit is assumed to be running and loaded on its Class 1E bus at least two hours before the onsite arrival of winds in excess of 73 mph. Therefore, the common cause failure parameters were adjusted to remove one EDG train from the start (diesel and ventilation fan), to run the first hour (diesel), and to close the output breakers for each unit. In response to RAI 19.01-31, this specific compensatory action and others credited in the risk evaluations are documented in FSAR Section 19.4.6, “ABWR Shutdown Risk,” and COM 19.4-1 CR 10-15528 Action 2. These specific actions include developing an abnormal operating procedure for severe weather that is consistent with NUMARC 87-00, Revisions 1, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,” Initiative 2, “Procedures,” and Section 2.11, “Hurricane Preparations.” This abnormal operating procedure will contain specific requirements as listed below:

- Action shall be initiated to place the units in Mode 3 (Hot Shutdown) at least 2 hours before wind speeds exceed 73 mph (or 96 mph as determined by discussions with the Transmission Distribution Service Provider (TDSP)). The applicability of this requirement is for units in Modes 1 and 2. Units in Modes 3, 4, or 5 will not be changed/maintained in Modes 3, 4, or 5.

- One EDG in each unit is started and loaded onto its safety bus, and the bus is disconnected from offsite power at least 2 hours before the onsite arrival of winds in excess of 73 mph.

- If an unstable electrical grid develops or is predicted by the TDSP, the remaining diesel generators are started and loaded on their safety buses, and the buses are disconnected from offsite power.

- If applicable for the current unit mode, the RCIC will be verified as available to provide core cooling in the event of a station blackout.

- The portable diesel driven fire pump will be staged in an onsite Seismic Category I structure prior to arrival onsite of sustained winds in excess of 73mph.

- If the containment is inerted at the time of the hurricane warning, it will remain inerted during a forced shutdown due to a hurricane and in anticipation of restoring the units to operation after the hurricane has passed.

Based on (a) results of the quantitative assessment and sensitivity analysis that satisfy the requirements of 10 CFR 52.79(d)(1); (b) COM 19.4-1 to develop abnormal operating procedures for hurricanes that will contain the specific requirements documented in FSAR Section 19.4.6; and (c) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-9 (RAI 19.01-31) are resolved. Verification that the applicant's proposed changes are in the revised FSAR sections is being tracked as **Confirmatory Item 19-15**.

19Q.7.6 Success Criteria

Under the subheading "(1) Decay Heat Removal from RPV," the applicant provides the following:

- STD DEP 5.4-1 Reactor Water Cleanup System

The STP plant-specific design includes one pump that provides 100 percent capacity instead of the two pumps operating at 50 percent capacity in the original ABWR design. See the discussion in Section 19Q.4.1.

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

The RHR design in the STP design will have three RHR loops connected to the FPCCS instead of the two RHR loops in the original ABWR design. See the discussion in Section 19Q.4.2.

19Q.7.7.1 Loss of RHR Due to Failure in the Operating RHR System

- STD DEP 10.4-5 Condensate and Feedwater System

The applicant states that if all RHR systems fail, the RPV will pressurize and the main condenser can become available by (1) opening the MSIVs; (2) drawing a vacuum in the condenser; and (3) operating the FW, condensate booster, and condensate pumps for makeup.

Table 19Q-1 references an updated list of features in STP Units 3 and 4 that minimize the risk of a shutdown by incorporating certain design departures and site parameters. In Table 19Q-2, there is an updated list of success criteria for incorporating certain design departures to prevent core damage. The staff found these changes acceptable.

Loss of RHR in Mode 3 or 4

- STD DEP 5.4-1 Reactor Water Cleanup System

For the RWCU, the STP plant-specific design uses a single pump providing 100 percent capacity instead of the two pumps operating at 50 percent capacity in the original ABWR design. See the discussion in Section 19Q.4.1.

Loss of RHR in Mode 5

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

The RHR design in the STP design will have three RHR loops connected to the FPCCS instead of the two RHR loops in the original ABWR design. See the discussion in Section 19Q.4.2.

Additional Design Departure

Regarding STD Departure 7.7-1 on RPV water level instrumentation, the DCD states that all instrument lines are flushed even when they do not need to be. The STP design addresses condensable gas buildup in the reactor vessel-referenced leg water level instrumentation by using the CRD to continually flush the instrument lines. NRC staff recognized that the CRD system may not be operating in Modes 4 and 5 and is not required to operate in Modes 4, 5, and 6, according to the TS. Therefore, the staff requested the applicant to address, in the STP Departure documentation and in the DCD documentation, how the instrument lines will be flushed during Modes 4 and 5 and how this action will be controlled (e.g., COL action item, etc.).

The staff then issued RAI 19-16 requesting the applicant to describe how the instrument lines will be flushed during Modes 4 and 5. In the applicant's response dated December 3, 2009 (ML093421266), the Standard Departure, STP DEP 7.7-1 was editorially changed to address those instrument lines with a condensing chamber that will be continually flushed by the CRD system. The applicant also discussed GL 92–04 and IN 93-27 which covered observed degassing in the BWR reference legs during cooldown and depressurization of operating BWRs. The applicant explained that once the plant enters Mode 4 and continues into Mode 5, the reactor is sub-cooled and fully depressurized. As a result, during Mode 4 and Mode 5, degassing in the reactor vessel reference legs is not of concern, and there is no need for continued supply of reference leg purge from the CRD system. In addition, the Shutdown Level and Reactor Well Indications do not require purge flow from the CRD system. The staff found the response acceptable and agrees that there is no effect on the shutdown PRA from this departure. Therefore RAI 19-16 is resolved.

19Q.5 Post Combined License Activities

The applicant identifies commitments (COM 19.9-10, COM 19.9-22, COM 19.9-23, COM 19.9-24, and COM 19.9-27) to address COL License Information Items 19.9, 19.19d, 19.19e, 19.19f, and 19.19j as discussed in SER Section 19.9.4.

19Q.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to “ABWR Shutdown Risk Assessment.” With the exception of **Confirmatory Item 19-15**, 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 “ABWR Shutdown Risk Assessment” that were incorporated by reference have been resolved. In addition, based on the above discussion on the “ABWR Shutdown Risk Assessment,” the staff concluded, pending completion of Confirmatory Item 19-15, that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19QA Fault Trees

Appendix 19QA of the STP Units 3 and 4 COL FSAR incorporates by reference with no departures or supplements Appendix 19QA, “Fault Trees,” of Revision 4 of the ABWR DCD, which is itself incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this appendix remains for review.[1] The staff’s review confirmed that there is no outstanding issue related to this appendix. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to “Fault Trees” of the ABWR Design have been resolved.

19QB DHR Reliability Study

19QB.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19QB of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

19QB.2 Summary of Application

Appendix 19QB of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19QB of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19QB, the applicant provides the following:

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 5.4-1 Reactor Water Cleanup System

This departure addresses the STP plant-specific design change from requiring both pumps, as in the ABWR DCD design, to a single CUW pump operating at 100 percent capacity during operating Modes 4 and 5.

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.

19QB.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19QB.4 Technical Evaluation

NRC staff reviewed Appendix 19QB of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "DHR Reliability Study."

The staff reviewed the information in the COL FSAR:

Tier 2 Departures Not Requiring Prior NRC Approval

19QB.5 Decay Heat Removal Capability of CUW and FPC

- STD DEP 5.4-1 Reactor Water Cleanup System

In the STP plant-specific design, a single CUW pump is needed to operate and provide 100 percent capacity during operating Modes 4 and 5. This design is a change from the requirement of both pumps in the ABWR original design. NRC staff agreed that this design change represents an improvement in the reliability of the CUW system and a reduction in the risk of a shutdown. The CUW can mitigate a loss of DHR after 8 days post-shutdown.

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

19QB.5 Post Combined License Activities

There are no post COL activities related to this section.

19QB.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 NRC staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to "DHR Reliability Study" that were incorporated by reference 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.

The staff found it reasonable that the identified Tier 2 departures are characterized as not requiring prior NRC approval per 10 CFR Part 52, Appendix A, Section VIII.B.5. In addition, the staff concluded that the relevant information in the COL FSAR is acceptable and meets the requirements defined in the ABWR DCD.

19QC Review of Significant Shutdown Events: Electrical Power and Decay Heat Removal

19QC.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19QC of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

19QC.2 Summary of Application

Appendix 19QC of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19QC of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in COL FSAR Section 19QC.1, the applicant provides the following:

Supplemental Information

The applicant provides supplemental information concerning the review of Electric Power Research Institute (EPRI) Topical Report (TR)-1003113, "An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989–2000)."

19QC.3 Regulatory Basis

The relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19QC.4 Technical Evaluation

NRC staff reviewed Appendix 19QC of the STP Units 3 and 4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to the "Review of Significant Shutdown Events: Electrical Power and Decay Heat Removal."

The staff reviewed the information in the COL FSAR:

19QC.1 Review of Significant Shutdown Events

A review of EPRI TR-1003113, "An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989-2000)," provides additional information of more recent

[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.

shutdown operating experience. However, the information does not identify any new or unique challenges to shutdown safety that are not identified in the referenced ABWR DCD. The staff found this updated review of significant shutdown events acceptable.

19QC.5 Post Combined License Activities

There are no post COL activities related to this section.

19QC.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 NRC staff's review confirmed that the applicant has addressed the required information, and no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and 10 CFR Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to "Review of Significant Shutdown Events: Electrical Power and Decay Heat Removal" that were incorporated by reference have been resolved.

19R Probabilistic Flooding Analysis (Related to RG 1.206, Part I, C.I.19, Appendix A, 19.1.4.1.1, "Description of the Level 1 PRA for Operations at Power"; 19.1.4.1.2, "Results from Level 1 PRA for Operations at Power"; and 19.1.5.3, "Other External Events.")

19R.1 Introduction

This FSAR appendix described the text changes and supplemental information in Appendix 19R of the ABWR DCD due to the departures of the STP Unit 3 and 4 design from those described in the ABWR DCD.

19R.2 Summary of Application

Appendix 19R of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 19R of the ABWR DCD, Revision 4 referenced in 10 CFR Part 52, Appendix A.

In addition, in FSAR Appendix 19R, the applicant provides the following:

Tier 1 Departures

- STP DEP T1 5.0-1 Site Parameters

This departure identifies increases in the site design-basis flood level from 30.5 centimeters below grade to 182.9 centimeters above the grade level. To protect the safety-related SSCs, this departure replaces the exterior doors of the reactor building and the control building located below the maximum flood elevation with watertight doors.

Tier 2 Departures Not Requiring Prior NRC Approval

- STP DEP 10.4-2 Main Condenser

This departure describes how STP Units 3 and 4 will utilize three condenser shells cross-connected to equalize the pressure, with each shell containing four tube bundles and parallel circulating water flow. This departure provides four 25 percent capacity circulating water pumps discharging into a common header.

- STP DEP 1.2-2 Turbine Building

This departure addresses why the turbine generator described in the referenced ABWR DCD is now obsolete and how the replacement will differ dimensionally. The turbine cycle equipment, such as FW heaters and pumps, also differs from the cycle equipment described in the ABWR DCD. This departure replaces the power generation heat sink described in the DCD (natural draft cooling tower) with a cooling reservoir, and the design now includes condensate booster pumps. Also, a dual-voltage design that requires the relocation of major components into and within the turbine building replaces the MVES design.

- STP DEP 9.2-10 Turbine Service Water System (Table 19R-1)

This departure addresses the changes to the TSW that include the TSW pump head and discharge flow, the TSW system design pressure, the location of the TSW pump house, the temperature increase and pressure drop across the turbine cooling water (TCW) heat exchangers, and the number of TCW discharge lines. A filling line is also added to the TSW pump discharge, and the TSW system inlet and outlet are modified to reflect that these lines come from and go to the main cooling reservoir.

- STP DEP 19R-1 Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

This departure addresses the internal flooding of the control building from the elimination of vacuum breaker valves on the supply and return piping connecting to the RBCW heat exchangers. The elimination of the vacuum breaker valves is due to the RSW system design changes that include the use of horizontal-type pumps instead of vertical wet-pit type pumps and piping configuration changes between the UHS basin and control building.

Supplemental Information

Contents in Sections and Subsections 19R.1, 19R.3, 19R.4, 19R.4.2.4, 19R.4.2.5, 19R.4.3, 19R.4.4, 19R.4.6, 19R.5.3, 19R.5.4.1, 19R.5.2, 19R.5.6, 19R6.1, 19R6.2, 19R.6.4, and 19R.6.6 are also revised. There is also a new Section 19R.7 for the STP Units 3 and 4 plant-specific analysis.

The applicant provides site-specific supplemental information in the following sections of the STP Units 3 and 4 COL FSAR. This supplemental information presents the analysis performed for RSW pump house internal flooding and also addresses departure STP DEP 19R-1 "Internal Flooding Due to Removal of RSW Vacuum Breaker Valves":

- Section 19R.1 "Introduction and Summary"
- Section 19R.3 "Screening Analysis - Water Sources and Buildings"
- Section 19R.4 "Deterministic Flood Analysis")
- Subsection 19R.4.2.4 "Watertight Doors"
- Subsection 19R.4.2.5 "Floor Drains"
- Section 19R.4.6 "RSW Pump House"
- Section 19R.5.2 "Methodology"
- Section 19R.5.6 "RSW Pump House"
- Section 19R.6.1 "Results"

- Section 19R.6.2 "Insights Gained from Analysis"
- Section 19R.6.4 "Operator Actions"
- Table 19R-1 "Sources of Water"
- Table 19R-7 "ABWR Features to Prevent/Mitigate Flooding"

The applicant provides site-specific supplemental information in the following sections of the STP Units 3 and 4 COL FSAR. This supplemental information updates the analysis performed for control building internal flooding to address Departure STP DEP 19R-1.

- Section 19R.1 "Introduction and Summary"
- Section 19R.4.4 "Control Building"
- Section 19R.5.4.1 "RSW Line Breaks"
- Section 19R.6.2 "Insights Gained from Analysis"
- Table 19R-1 "Sources of Water"
- Table 19R-7 "ABWR Features to Prevent/Mitigate Flooding"

The applicant provides site-specific supplemental information in the following sections of the STP Units 3 and 4 COL FSAR. This supplemental information updates the analysis performed for turbine building internal flooding to address Departures STP DEP 1.2-2 ("Turbine Building"), STP DEP 10.4-2 ("Main Condenser"), and STP DEP 9.2-10 ("Turbine Service Water System").

- Section 19R.4.3 "Turbine Building Features"
- Section 19R.5.3 "Turbine Building"
- Section 19R.6.4 "Operator Actions"
- Section 19R.6.6 "Conclusions"
- Table 19R-1 "Sources of Water"
- Table 19R-6 "Internal Flooding Core Damage Frequency (CDF)"
- Figure 19R-7 "Turbine Building Flooding (Low PCHS)"

The applicant provides site-specific supplemental information in the following sections of the STP Units 3 and 4 COL FSAR. This supplemental information presents the analysis performed for external flooding and also addresses Departures STP DEP T1 5.0-1 ("Site Parameters") and STP DEP 1.2-2.

- Subsection 19R.4.2.4 "Watertight Doors"
- Section 19R.7 "External Flooding Evaluation"
- Figure 19R-6 "Reactor Building Arrangement - Elevation 12300 mm (1F)"

19R.3 Regulatory Basis

The relevant requirements for the Commission’s regulations, and the associated acceptance criteria, for reviewing supplemental information to support the COL application are described in Section 19.1.3 of this SER.

19R.4 Technical Evaluation

NRC staff reviewed Appendix 19R of the STP Units 3 and 4 COL FSAR. The staff checked the referenced DCD to ensure that the combination of the DCD and the information in the COL 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 "Probabilistic Flooding Analysis."

The staff reviewed the information in the COL FSAR:

Tier 1 Departures .

- STP DEP T1 5.0-1 Site Parameters

NRC staff reviewed the supplemental information in Appendix 19R.7 of the STP Units 3 and 4 COL FSAR related to the external flooding analysis (STP DEP T1 5.0-1). This supplemental information describes the probabilistic external flooding analysis and provides the results and risk insights. The staff's findings from the review of this supplemental information include the following:

a. FSAR Section 19R.7,"External Flooding Evaluation," Revision 2, qualitatively describes the plant-specific PRA for external flooding due to multiple concurrent upstream dam failures. The staff issued RAI 19.01-11 requesting the applicant to provide the quantitative information associated with the plant-specific risk for external flooding due to these dam failures.

   The applicant's response to RAI 19.01-11 dated July 23, 2009 (ML092080083), states that the potential design-basis external flood has been reanalyzed in response to RAI 02.04.04-9, in Section 2.4S of the STP Units 3 and 4 COL FSAR. The new flood height associated with the nonmechanistic, multiple-cascading upstream dam failure scenario described in Chapter 2.4S is 32.5 ft MSL. With a wave run-up, the maximum water level from the multiple cascading dam failure is 34.4 ft MSL, which is below the openings to safety-related buildings at the STP Units 3 and 4 site. For this reason, this flood scenario is no longer considered a potential source of external flooding to be included in the site-specific PRA described in Appendix 19R. The applicant also states that Appendix 19R, Appendix 19Q, and Chapter 19.4 of the STP Units 3 and 4 COL FSAR will be modified accordingly.

   The staff found that the applicant's response to RAI 19.01-11 sufficiently addresses the concerns associated with this RAI. The staff confirmed that the STP Units 3 and 4 COL FSAR has been revised accordingly. Based on the above discussion, RAI 19.01-11 is resolved.

   FSAR Section 19R.7, Revision 2, qualitatively describes the plant-specific PRA for external flooding due to a main cooling reservoir breach. The staff issued RAI 19.01-10 requesting the applicant to provide the quantitative information associated with the plant-specific PRA for external flooding due to a main cooling reservoir breach.

   The applicant's response to RAI 19.01-10 dated July 23, 2009 (ML092080083) states that the main cooling reservoir breach evaluation results described in the STP Units 3 and 4 COL FSAR will not significantly affect the Level 1 results presented in the ABWR SSAR, if they were summed with the internal events results. In order to remain consistent with 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.

evaluations performed for other traditional external events (i.e., fire and seismic), the external flooding analyses were treated as screening evaluations and were not considered for inclusion with the Level 1 results discussed in the DCD. The important risk insights are incorporated into FSAR Chapter 19 where appropriate (e.g., watertight doors, operator training, etc.). The initiating event frequency for a main cooling reservoir breach is an estimated 1.0E-06 per year. The CDF for a main cooling reservoir breach is an estimated 1.1E-07 per year. The applicant also provides the basis for the initiating event frequency of a main cooling reservoir breach, in addition to the significant accident sequences leading to core damage. The most significant sequence (CDF of 1.0E-07 per year) includes a main cooling reservoir breach with an operator failure to close the control building watertight access door. The applicant adds that the detailed screening evaluation is available at the site for review by the staff. The staff found that the applicant's response to RAI 19.01-10 does not sufficiently address the concerns in this RAI. These concerns are addressed further during the staff's audit of the STP Units 3 and 4 PRA in September 2009. The staff issued RAI 19-30 to include the unresolved issues of RAI 19.01-10. Therefore, RAI 19.01-10 is considered resolved and closed.

During the staff's audit of the STP Units 3 and 4 PRA in September 2009, the staff reviewed STP's detailed screening evaluation for external flooding due to a main cooling reservoir breach. This evaluation is in the Engineering/Licensing Evaluation titled, "External Flooding Event, Breach of the Main Cooling Reservoir," dated April 20, 2009 (ML093560778). The staff issued RAI 19-30 requesting the applicant to justify (1) the site-specific main cooling reservoir breach frequency of 1.0E-6 per year, and (2) the reduction factors used to obtain this frequency from the generic dam failure frequency of 1E-4 per year. The staff also requested additional information on the probability (basic event - OCD) that the operator will fail to close the single, normally open, watertight access door between the service building and the control building. The staff's questions include:

(1) FSAR Section 2.4S.10, Revision 3, states: "All safety-related facilities in the power block are designed to be water tight at or below elevation 40.0 ft MSL. All water tight doors and hatches are normally closed under administrative controls and open outward. <u>A</u> main cooling reservoir <u>embankment breach near the STP Units 3 and 4 power block area would not provide sufficient time for implementation of emergency operating procedures or flood warning systems.</u> As all watertight doors and hatches are to remain in a closed position, no emergency operating procedures or plant Technical Specifications (plant shutdown), which are discussed in Subsection 2.4S.14, are required for implementation of flood protection measures." The main cooling reservoir external flooding PRA analysis described in Appendix 19R of the FSAR is not consistent with the above statement in that under Appendix 19R the water tight door between the service building and the control building is normally open and takes credit for emergency operating procedures and operator action to close this water tight door during main cooling reservoir breach. Please clarify this inconsistency and revise the FSAR as appropriate.

(2) In STP's response to RAI 19.01-10, STP stated that the overtopping, slope protection erosion, and sliding failure modes are not applicable to the main cooling reservoir design. Please justify why these failure modes are not applicable to the main cooling reservoir design, and provide the basis for the reductions in dam failure frequency as a result of excluding these failure

modes. In your discussion on why the main cooling reservoir cannot overtop, please include the following information:

- The maximum pumping capacity to the main cooling reservoir from the Colorado River and the maximum discharge capacity to the Colorado River.

- The frequency at which the main cooling reservoir levels are monitored and how this information is alarmed/displayed in the control room.

- The procedures used to control main cooling reservoir level, and the response procedures if main cooling reservoir level becomes too high.

(3) FSAR Appendix 19R.7.4.1, Revision 3, states: "A breach of the main cooling reservoir could occur suddenly or progress over many minutes." This section of the FSAR also discusses other dam breaches noting that the failure time of most breaches is 15 minutes to 1 hour, and some breaches become fully developed in as little as 6 minutes. A sudden breach of the main cooling reservoir (e.g., seismic liquidification) may not provide sufficient time for the operator to close the water tight door between the service building and the control building (i.e., basic event OCD = 1.0). Please address the external flooding analysis due to sudden main cooling reservoir breaches.

(4) Please assess the impact of Category 4 and 5 hurricanes on the frequency of main cooling reservoir breach. Address how a storm surge from such a hurricane would affect the main cooling reservoir levee system and the exterior side of the reservoir that has no liner.

(5) Please provide your data sources for dam failures that include infantile dam's failures that were used to support your reduction factor for satisfactory operation of the main cooling reservoir for five years. Based on staff review of dam failures from the National Performance of Dams Program (NPDP), developed by the Department of Civil and Environmental Engineering at Stanford University, including the Taum Sauk dam failure in 2005, the inclusion of infantile dam failures would result in generic dams break frequencies greater than 1E-4 per year. In addition, it appears that the reduction you credited for satisfactory operation of the main cooling reservoir seems to be double-counting. Please address these issues in your response.

(6) Please justify the factor of three reduction you used, based on the assumption that the location of a breach is limited to a thousand foot section. Please explain why any thousand foot section in the 16,250 foot perimeter facing the safety related buildings can not cause a flood.

(7) Please assess the impact of a main cooling reservoir breach during cold shutdown and refueling if secondary and primary containment has open penetrations to facilitate maintenance. Please consider the elevations of these penetrations in your assessment.

(8) Please document if the assumptions, insights, or conclusions in the referenced calculation change given the revised main cooling reservoir breach evaluation in Section 2.4.4.1.2 of the FSAR.

(9) The staff needs more information on the probability (basic event- OCD) of the operator failing to close the single normally open flood door between the service building and the control building. To justify the human error probability 0.1, please provide the following information:

- The criterion that you will supply to the guard at security house to determine if the main cooling reservoir has breached.
- The process by which these procedures will be controlled.
- The potential for ambiguous visual indication on the occurrence of a main cooling reservoir breach including: the occurrence of local ponding due to heavy rains and the ability of the guard to identify increased flood levels due to reduced visibility during heavy rain storms, fog, etc., particularly at night time.
- Appendix 19R.7.5.1 of the FSAR states: "…a minimum available warning time from water at the South Security Gate House, approximately El. 32.0' MSL, to water at the entrances to safety-related buildings, El. 35.0' MSL. At least 30 minutes is available for operator action to close the normally open access door between the Service Building and the Control Building once water reaches the South Security Gate House." Please sufficiently justify the operator action time of at least 30 minutes.

RAI 19-30 was tracked as Open Item 19-12 in the SER with open items. In response to Open Item 19-12, the applicant submitted the final response to RAI 19-30 on July 28, 2010 (ML102110184). This RAI response changed the status for all watertight doors and hatches to be normally closed. This change in door status will be documented in FSAR Section 2.4S.10, “Flooding Protection Requirements.” This RAI response also used the ASME/ANS RA-Sa-2009, Section 6-2.3, “The Fundamental Criteria for Screening External Events Other Than Fire and Seismic Events.” Criterion (a) was used to screen external flood scenarios from detailed quantitative evaluation. Criterion (a) is satisfied since the STP design for safety-related systems, structures, and components satisfies the requirements of SRP Section 3.4.2, Revision 3, which was in effect at the time of the COL application. Based on the proposed FSAR change in watertight door status and the proposed removal of the screening quantification for a postulated main cooling reservoir breach, the following COL FSAR sections will be revised:

- FSAR Section 2.4S.10, “Flooding Protection Requirements”
- FSAR Section 2.4S.14, “Technical Specifications and Emergency Operation Requirements”
- FSAR Section 19.4.5, “ABWR Probabilistic Flooding Analysis”
- FSAR Subsection 19.8.5.3, “Features Selected”
- FSAR Section 19.9.3, “Event Specific Procedures for Severe Accident Flooding”
- FSAR Section 19.11, “Human Action Overview”
- FSAR Section 19K.10, “Identification of Important Capabilities Outside the Control Room”

- FSAR Section 19R.6.4, “Operation Actions”

- FSAR Section 19R.7, “External Flooding Evaluation”

Based on (1) the change in watertight door status to be normally closed, (2) the removal of the screening quantification for postulated main cooling reservoir breach, and (3) the proposed revisions to the affected COL FSAR sections, the staff concluded that the issues associated with Open Item 19-12 (RAI 19-30) have been resolved. The staff confirmed that the proposed revisions are incorporated into Revision 4 of the FSAR. Therefore, the staff found the applicant’s modeling of external floods acceptable.

Tier 2 Departures Not Requiring Prior NRC Approval

- STP DEP 19R-1 Internal Flooding Due to Removal of RSW Vacuum Breaker Valves

NRC staff reviewed the supplemental information in Appendix 19R of the STP Units 3 and 4 COL FSAR related to the control building internal flooding analysis. This supplemental information updates the probabilistic internal flooding analysis for the control building to address Departure STP DEP 19R-1. The staff's findings from the review of this supplemental information are as follows:

Departure STP DEP 9.2-5 is associated with Revision 2 of the STP Units 3 and 4 COL FSAR. The departure increases the RSW flow rate per pump from 1,800 m^3/h to 3,290 m^3/h and also increases the RSW pipe sizes. This change can impact the plant-specific PRA for control building flooding in FSAR Section 19R.5.4, "Control Building," Revision 2. For example, this departure can impact the timing associated with operator actions in top events "OPACT1," "OPACT2," and "OPACT3" in the event tree for control building flooding due to an RSW line break (refer to Figure 19R-9, "RSW Control Building Flood," in the ABWR SSAR). In addition, the departures that were considered in the internal events PRA (e.g., STD DEP T1 2.4-3, STD DEP T1 3.4-1, STD DEP 8.3-1, STP DEP 9.2-5, and STD DEP 19.3-1) can impact the failure probabilities associated with the top events for bringing the reactor to a safe shutdown condition in the control building flooding event tree. The staff issued RAI 19.01-6 requesting the applicant to describe the risk impact that the departures have on the PRA results for control building flooding.

The applicant’s response to RAI 19.01-6 dated July 23, 2009 (ML092080083), states that the RSW pump flow rates do not directly affect the computed leakage from the postulated RSW pipe failure, as this leakage is based only on the operating pressure within the pipe, the pipe crack size, and the volume of the RSW piping, which contributes to the flood source. Larger pipe diameters are offset by the reduced amount of piping associated with the redesigned RSW system. Because the break size associated with the increased pipe diameter is bound by the size assumed in the DCD, and the increased flow rate of the RSW pumps does not affect the flow rate out of the break, there is no significant effect on operator timing and no change to the PRA described in the DCD. The revised water volume in the control building basement from the RSW pipe failure described in Appendix 19R is approximately 6,500 ft^3 (~184 m^3), with automatic isolation. This volume results in a water level of 7.6 ft (~2.3 m), which is well below the 5-m maximum of the RSW design description in Tier 1, Section 2.11.9. The lower result is due to the significantly shorter length of the RSW pipe that drains into the RCW pump room from the RSW system following an RSW train isolation and draindown. The departures that were considered in the internal events PRA do not significantly affect the PRA results described in the DCD, as indicated in Chapter 19.3, so there is no required change to control building flooding from these departures under RG 1.206, C.III.I.19.

The staff found that the applicant's response to RAI 19.01-6 sufficiently addresses the concerns associated with this RAI. Based on the above discussion, RAI 19.01-6 is resolved. The STP departures do not significantly affect the PRA results for control building internal flooding.

- STP DEP 1.2-2 Turbine Building
- STP DEP 10.4-2 Main Condenser
- STP DEP 9.2-10 Turbine Service Water System

NRC staff reviewed the supplemental information in Appendix 19R of the STP Units 3 and 4 COL FSAR related to the turbine building internal flooding analysis. This supplemental information updates the probabilistic internal flooding analysis for the turbine building to address Departures STP DEP 1.2-2, STP DEP 10.4-2, and STP DEP 9.2-10. The staff's findings from the review of this supplemental information are as follows:

Departure STP DEP 10.4-2 increases the number of circulating water pumps to four. This increase can impact the PRA for turbine building flooding in Section 19R.5.3, "Turbine Building," of the STP FSAR Revision 2. For example, this departure can impact the failure probabilities associated with top events "PTRIP" and "VCLOSE" in the turbine building flooding event tree (refer to Figure 19R-8, "Turbine Building Flooding, High PCHS," in the ABWR SSAR). In addition, the departures that were considered in the internal events PRA (e.g., STD DEP T1 2.4-3, STD DEP T1 3.4-1, STD DEP 8.3-1, STP DEP 9.2-5, and STD DEP 19.3-1) could impact the failure probabilities associated with the top event for bringing the reactor to a safe shutdown condition in the turbine building flooding event tree. The staff issued RAI 19.01-5 requesting the applicant to describe the risk impact from the departures on the PRA results for turbine building flooding.

The applicant's response to RAI 19.01-5 dated July 23, 2009 (ML092080083), states that the response of the plant to a failure of the main circulating water piping assumes that even if the automatic protection does not work, the water will exit the turbine building through the truck doors, according to DCD Appendix 19R.1:

> In the unlikely event this automatic protection fails and the operator fails to take any action, potential flood waters would still be prevented from reaching the service building. Potential flood waters would be expected to exit the turbine building through the non-watertight truck entrance door.

Also, increasing the number of circulating water pumps does not affect the level setpoints at which the circulating water pumps trip and the pump isolation and condenser isolation valves close, or the plant's response to a circulating water flooding event. Therefore, as described in the STP Units 3 and 4 COL FSAR, there is no change to the PRA results in the DCD. The top event "PTRIP" in the turbine building flooding (High PCHS) event tree, (SSAR Figure 19R-8), has no branch in the event tree for the High PCHS design because tripping the circulating water pumps does not stop the circulating water flow, and is therefore unaffected by the number of circulating water pumps in the circulating water system. The top event "VCLOSE" is also unaffected by the changes associated with STP DEP 10.4-2. The function modeled by the "VLCOSE" includes the condenser isolation valves, one for each condenser element, and the circulating water pump isolation valves. The value in SSAR Figure 19R-8 derived from the data in SSAR Table 19R-4 represents the failure of one of three isolation valves (condenser isolation valves) and the CCF with any pump isolation valve represented by the beta factor in Table 19R-4. There is no change to the modeling of the turbine building flooding event tree in Figure 19R-8 of the SSAR. The departures that were considered in the internal events PRA do not significantly affect the PRA results described in the DCD and in Chapter 19.3, so there is no required change to turbine building flooding from these departures under RG 1.206, C.III.I.19.

The staff found that the applicant's response to RAI 19.01-5 sufficiently addresses the concerns associated with this RAI. Based on the above discussion, RAI 19.01-5 is resolved.

Supplemental Information

NRC staff reviewed the supplemental information in Appendix 19R of the STP Units 3 and 4 COL FSAR related to the internal flooding analysis of the RSW pump house. This supplemental information describes the probabilistic and deterministic internal flooding analysis for the RSW pump house and provides the results and risk insights. The staff's findings from the review of this supplemental information are as follows:

a. FSAR Appendix 19R.5.6 ("RSW Pump House") states:

   Unisolated breaks in the fire water system could cause inter-divisional flooding since the RSW divisional separation splits the RSW pump house into three, watertight compartments.

   However, Appendix 19R, of the STP FSAR does not provide or describe a PRA for internal flooding due to unisolated breaks in the fire water system in the RSW pump house. The staff issued RAI 19.01-7 requesting the applicant to describe the PRA internal flooding analysis for this scenario.

   The applicant’s response to RAI 19.01-7 dated July 23, 2009 (ML092080083), states that floods associated with fire water system leaks and piping failures and usage in the RSW pump house are less significant than a flood from the RSW piping, as described in Section 19R.1 of the STP Units 3 and 4 COL FSAR, because of lower water flows and external water isolation capability. If analyzed, fire water floods would be bounded by the results of the RSW piping floods, which are included in Appendix 19R.

   The staff found that the applicant's response to RAI 19.01-7 sufficiently addresses the concerns associated with this RAI. Based on the above discussion, RAI 19.01-7 is resolved.

   FSAR Subsection 19R.5.6.1, "RSW Line Breaks," qualitatively describes the plant-specific PRA for internal flooding due to RSW line breaks in the RSW pump house. The staff issued RAI 19.01-8 requesting the applicant to provide the quantitative information associated with the plant-specific PRA for internal flooding due to RSW line breaks in the RSW pump house.

   The applicant’s response to RAI 19.01-8 dated July 23, 2009 (ML092080083), states that there was a screening evaluation consistent with that of the ABWR DCD and SSAR. The evaluation used the PRA information in Appendix 19R of the SSAR and resulted in a very small change in total CDF, when compared to the SSAR internal events results. The total CDF for this event from the screening assessment is 3.8E-08 per year. The applicant’s response also describes in detail the screening evaluation including the assumptions, significant accident sequences and their mean CDFs, initiating event frequency estimates and their basis, and the top event failure probabilities and their basis.

The staff found that the applicant's response to RAI 19.01-8 sufficiently addresses the concerns associated with this RAI. Based on the above discussion, RAI 19.01-8 is resolved.

19R.5 Post Combined License Activities

- The applicant identifies commitments (COM 19.9-3 and COM 19.9-9) to address COL License Information Items 19.3 and 19.10 9 as discussed in SER Section 19.9.4.

19R.6 Conclusion

NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant has addressed the required information relating to "Probabilistic Flood Analysis." 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 "Probabilistic Flood Analysis" that were incorporated by reference have been resolved.

In addition, based on the above discussion on the "Probabilistic Flood Analysis," the staff concluded that the relevant information in the COL FSAR is acceptable and meets the applicable requirements described in Section 19.1.3 of this SER.

19S <u>Aircraft Impact Assessment</u>

This appendix of the FSAR discusses the design features and functional capabilities of the ABWR design to mitigate the effects of a large, commercial aircraft in accordance with 10 CFR 50.150. This appendix also describes how the identified design features and functional capabilities show that, with reduced use of operator actions, the reactor core remains cooled, or the containment remains intact, and the spent fuel cooling or spent fuel pool integrity is maintained.

Appendix 19S of the STP Units 3 and 4 COL FSAR incorporates by reference, with no departures or supplements, Appendix 19S, "Aircraft Impact Assessment" of Revision 3 of the application to amend the design certification rule for the U. S. advanced boiling water reactor to implement 10 CFR 50.150. NRC staff reviewed the application and checked the referenced DCD amendment application to ensure that no issue relating to this section remains for review. The staff's review confirmed that there is no outstanding information outside of the DCD amendment related to this section. If the proposed amendment is issued as a final rule, all nuclear safety issues relating to the aircraft impact assessment will be resolved pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Sections VI.B.1, 2, and 3.

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GE Nuclear Energy

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