6.0 ENGINEERED SAFETY FEATURES

This chapter of the Final Safety Analysis Report (FSAR) discusses the design and functional requirements of engineered safety features (ESFs) of the plant that are provided to mitigate the consequences of postulated accidents. The ESF consist of containment systems, core cooling systems, habitability systems, and fission product removal and control systems.

6.1 Engineered Safety Feature Materials

Materials used in the ESF components have been evaluated to ensure that material interactions do not occur that can potentially impair the operation of the ESFs. Materials have been selected to withstand the environmental conditions encountered during normal operations and during any postulated loss-of-coolant accident (LOCA). Their compatibility with core and containment spray solutions has been considered, and the effects of radiolytic decomposition products have been evaluated.

6.1.1 Metallic Materials

6.1.1.1 *Introduction*

This section of the FSAR addresses materials selection, fabrication, processing, and compatibility with ESF fluids, components, and system cleaning and composition of thermal insulation in ESF systems.

6.1.1.2 *Summary of Application*

Section 6.1.1 of the South Texas Project (STP) Units 3 and 4 FSAR incorporates by reference Section 6.1.1 of the certified Advanced Boling-Water Reactor (ABWR) design control document (DCD), Revision 4, referenced in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Appendix A, with no departures.

In addition, the applicant provides the following:

Supplemental Information

In FSAR Subsection 6.1.1.1.1, "Material Specification," the applicant commits (COM 6.1-1) to provide site-specific information identified in ABWR DCD Table 6.1-1. However, in response to U.S. Nuclear Regulatory Commission (NRC) staff's request for additional information (RAI) 06.01.01-1, the applicant deletes Commitment (COM) 6.1-1 and provides in FSAR Table 6.1-1 the site-specific information identified in ABWR DCD Table 6.1-1. This issue is discussed below.

6.1.1.3 *Regulatory Basis*

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

In addition, the relevant requirements for the Commission's regulations, and the associated acceptance criteria, for reviewing supplemental information are in Section 6.1.1 of

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” the Standard Review Plan (SRP).

6.1.1.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.1.1 of the certified ABWR DCD. The staff reviewed Section 6.1.1 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

In addition, NRC staff reviewed the supplemental information from the applicant related to the materials selection for the reactor building's cooling water system and reactor service water system components, which are considered site-specific information in ABWR DCD Table 6.1-1, as summarized below.

<u>*Supplemental Information*</u>

The applicant commits to address the following site-specific information identified in referenced ABWR DCD Table 6.1-1:

> Materials to be used in the Reactor Building Cooling Water System heat exchanger and the Reactor Service Water System pump and valves will be provided in the FSAR in accordance with 10 CFR 50.71(e) prior to the initiation of the respective unit preoperational testing. (COM 6.1-1).

In order for NRC staff to complete the review, the staff issued RAI 06.01.01-1 requesting the applicant to modify COL FSAR Table 6.1-1 to include material specifications and grades for the reactor building cooling water system heat exchangers and reactor service water system pumps, valves, and piping.

The applicant's response to RAI 06.01.01-1 in a letter dated January 28, 2010 (ML100330402**)** proposes to revise FSAR Subsection 6.1.1.1.1 and Table 6.1-1. The proposed revision to Subsection 6.1.1.1.1 states that the materials to be used in the reactor building cooling water system heat exchanger and the reactor service water system pump, piping, and valves are identified in FSAR Table 6.1-1. The revision to Subsection 6.1.1.1 also deletes COM 6.1-1, because this information will be in STP FSAR Revision 4 and not at a later date after the issuance of the COL, as originally proposed by the applicant. The applicant's proposed revision to Table 6.1-1 lists material specifications and grades for the reactor building cooling water system heat exchanger and the reactor service water system pump, piping, and valves. The heat exchanger plate material is titanium SB-265 Grade 1. No tube material is listed for the heat exchanger because the applicant intends to use an alternate plate-type heat exchanger, as permitted in ABWR DCD Subsection 9.2.11.2. Reactor service water system pumps and valves are fabricated from austenitic stainless castings and forgings that meet SA-351 Grades CF3M,

[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.

CF8, CF8M, and SA-182 Grade F316L specifications. Piping material is austenitic stainless steel and meets specifications SA-312 Grade TP 316 L and SA-358 Grade 316L.

The staff reviewed the applicant's response and found these material specifications and grades acceptable because they meet American Society of Mechanical Engineers (ASME) Code Section III requirements. The materials are also acceptable for their intended use because they are compatible with their operating environment and are resistant to stress corrosion cracking and erosion/corrosion. Therefore, the staff found the materials consistent with the guidance in Section 6.1.1 of the SRP (NUREG–0800). The staff verified that Revision 4 of FSAR Subsection 6.1.1.1.1 and Table 6.1-1 reflect the changes discussed in the response to RAI 06.01.01-1. RAI 06.01.01-1 is therefore resolved.

6.1.1.5 *Post Combined License Activities*

There are no post COL activities related to this section.

6.1.1.6 *Conclusion*

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

The staff's review confirmed that the COL applicant has adequately addressed the site-specific information identified in ABWR DCD Table 6.1-1, in accordance with Section 6.1.1 of NUREG-0800.

6.1.2 Organic Materials

6.1.2.1 *Introduction*

This section of the FSAR addresses the protective coating systems and organic materials used inside the containment. Evaluations are performed to ensure that the protective coatings will not fail under design-basis accident (DBA) conditions, and the materials will not generate an amount of solid debris that would impair the performance of the ESF systems. The performance of the protective coatings and organic materials should be examined for radiation and chemical effects in the containment.

6.1.2.2 *Summary of Application*

Section 6.1.2 of the STP Units 3 and 4 FSAR incorporates by reference Section 6.1.2 of the certified ABWR DCD, Revision 4, with no departures.

In addition, in FSAR Section 6.1.3, the applicant provides the following:

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

- COL License Information Item 6.1 Protective Coatings and Organic Materials

The applicant provides additional information to address this COL license information item. The applicant commits (COM 6.1-2) to analyze any containment coatings that do not comply with the guidance of Regulatory Guide (RG) 1.54 and American National Standards Institute (ANSI) N101.2 after the procurement of the components.

6.1.2.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is documented in NUREG-1503. The regulatory basis for reviewing the COL license information item is in Section 6.1.2 of NUREG-0800.

6.1.2.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.1.2 of the certified ABWR DCD. The staff reviewed Section 6.1.2 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

The staff also reviewed the conformance of Section 6.1.2 of the STP Units 3 and 4 COL FSAR with the guidance in RG 1.206 Chapter C.III.1, Section C.I.6.1.2, "Organic Materials." The staff used SRP Section 6.1.2 as guidance for reviewing the information in the FSAR to resolve COL License Information Item 6.1.

The staff reviewed the information in the COL FSAR:

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

- COL License Information Item 6.1 Protective Coatings and Organic Materials

NRC staff reviewed the resolution to COL License Information Item 6.1 related to the amount of unqualified coatings inside the containment and the generation rate of combustible gases from organic materials under DBA conditions. COL License Information Item 6.1 requires the COL applicant to provide the following specific information:

> The COL applicant shall:
>
> (1) Indicate the total amount of protective coatings and organic materials used inside the containment that do not meet the requirements of ANSI N101.2 and RG 1.54.
>
> (2) Evaluate the generation rate as a function of time of combustible gases that can be formed from organic materials under DBA conditions.

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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.

(3) Provide the technical basis and assumptions used for this evaluation (Subsections 6.1.2.1 and 6.1.2.2).

NRC staff reviewed the information from the applicant using the guidance described in Section 6.1.2 of the SRP (NUREG–0800) and RG 1.206. Because the applicant has proposed to provide the COL license information with a commitment (COM 6.1-2) following licensing, the purpose of the staff's review was to determine whether COM 6.1-2 adequately addresses the information required by the DCD.

The staff issued **RAI 06.01.02-1** requesting the applicant to state in the FSAR how the evaluation of these coatings and organic materials will be documented and retained in the plant's quality records, as part of the 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program.

In a letter dated July 7, 2009 (ML091900147), the applicant summarizes how the evaluation of combustible gas generation for the nonconforming coatings and organic materials in the containment will be performed in accordance with 10 CFR Part 50, Appendix B. The summary identifies key parameters needed to comply with the Appendix B program, including the technical basis of the evaluations, documentation, personnel qualifications, document revision controls, and records retention and audits. The applicant's response also proposes to modify FSAR Subsection 6.1.3.1 by adding the following closing sentence:

> The analysis will be documented and retained in plant quality records in accordance with applicable sections of 10 CFR Part 50, Appendix B.

The staff found this response acceptable because the 10 CFR Part 50, Appendix B requirement will be clearly stated in the FSAR. This issue was tracked as **Confirmatory Item 6.01.02-1** in the SER with open items. The staff verified that Revision 4 of FSAR Subsection 6.1.3.1 reflects the changes discussed in the response to RAI 06.01.02-1. RAI 06.01.02-1 is therefore resolved.

6.1.2.5 *Post Combined License Activities*

The applicant identifies the following commitment:

- Commitment (COM 6.1-2) – The inventory and analysis of nonconforming protective coatings and organic materials used inside the containment will be available to staff by the end of preoperational testing for each unit.

6.1.2.6 *Conclusion*

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

On the basis of the review of the STP COL application and the referenced DCD, the staff found that the applicant has adequately addressed COL License Information Item 6.1, in accordance with Section 6.1.2 of NUREG–0800.

6.2 Containment Systems

6.2.1 Containment Functional Design

The STP Units 3 and 4 containment system references the ABWR design certified by the NRC in March 1997. This design includes a containment structure as the primary containment, a secondary containment (reactor building) surrounding the primary containment, housing equipment essential to a safe shutdown of the reactor and fuel storage facilities, and supporting systems. The primary containment is designed to prevent the uncontrolled release of radioactivity into the environment, with a leakage rate of 0.5 percent by weight per day at the calculated peak containment pressure related to the DBA. The secondary containment is designed to confine the leakage of airborne radioactive materials from the primary containment. SSAR Figure 6.2.1 shows the principal features of the ABWR containment.

The certified containment functional design description and the NRC staff's evaluation of the design are in the ABWR FSER, NUREG–1503. The certified ABWR containment design is incorporated into the STP COL application by reference, except for the standard technical departures (STDs) noted and addressed in this section. In general, the departures consist of alternate analytical methodologies applied to the reconstitution of the containment DBA analyses.

6.2.1.1 *Introduction*

The discussion that follows briefly describes the features of the ABWR primary containment design.

A drywell consisting of two volumes, (1) an upper drywell (UD) volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system; safety/relief valves (SRVs); and the drywell heating, ventilation, and air conditioning (HVAC) coolers; and (2) a lower drywell (LD) volume housing the reactor internal pumps (RIPs), control rod drives (CRDs), and under-vessel components and servicing equipment.

The UD is a cylindrical, steel-lined, reinforced concrete structure with a removable steel head and a reinforced concrete steel diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the steel diaphragm floor, separates the LD from the wetwell. Ten UD-to-LD drywell connecting vents (DCVs), approximately 1 meter (m) x 2 m (3.3 feet [ft] x 6.6 ft) in cross sections, are built into the RPV pedestal. The DCVs extend downward through steel pipes with an inside diameter of 1.2 m (4 ft); each pipe has three horizontal vent outlets into the suppression pool.

The drywell, which has a net free volume of 7,350 m^3 (259,563 ft^3), is designed to withstand design pressure and temperature transients following a LOCA and the rapid reversal in pressure when the steam in the drywell is condensed by the emergency core cooling system (ECCS) flow during post-LOCA flooding of the RPV. A wetwell-to-drywell vacuum relief system will prevent backflooding of the suppression pool water into the LD and will protect the integrity of the steel diaphragm floor slab between the drywell and wetwell and the drywell structure and liner. The drywell design pressure and temperature are 310 kilopascal gauge (kPaG) (45 pounds per square inch gauge [psig]) and 171 °C (340 °F), respectively. The design drywell-to-wetwell differential pressures are +172.4 kPaG (25 psig) and -13.8 kPaG (-2 psig). The design drywell-to-reactor building negative differential pressure is -13.8 kPaG (-2 psig).

A system of drywell-to-wetwell vent channels will blow down from the drywell and discharge into the suppression pool following a LOCA. There are 30 vents in the vertical section of the LD below the suppression pool water level, each with a nominal diameter of 0.7 m (2.3 ft). These vents are arranged in 10 circumferential columns, each containing three vents. The three-vent centerlines in each column are located 3.5 m (11.48 ft), 4.87 m (15.98 ft), and 6.24 m (20.48 ft) below the suppression pool water level when the suppression pool is at the low water level. A wetwell consists of an air volume and suppression pool, with a net free-air volume of 5,960 m^3 (210,475 ft^3) and a minimum pool volume of 3,580 m^3 (126,427 ft^3) at the low water level.

The wetwell is designed for an internal pressure of 310 kPaG (45 psig) and a temperature of 103.9 °C (219 °F). The design wetwell-to-reactor building negative differential pressure is -13.8 kPaG (-2 psig). The suppression pool, which is located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell, is a large body of water that will serve as a heat sink for postulated transients and accidents and as a source of cooling water for the ECCS. In the case of transients that result in a loss of the ultimate heat sink, energy will be transferred to the pool by the discharge piping from the reactor system's SRVs. In the event of a LOCA in the drywell, the drywell atmosphere will be vented to the suppression pool through the system of drywell-to-wetwell vents.

This primary containment design basically uses combined features of the Mark II and Mark III designs, except that the drywell consists of UD and LD volumes. The vents to the suppression pool are a combination of the vertical Mark II and horizontal Mark III systems. The wetwell is similar to a Mark II wetwell.

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without a loss of function, the pressure and temperature conditions resulting from any postulated LOCA. Furthermore, the containment and enclosed structures must be designed to withstand a full range of loading conditions that are consistent with normal plant operations and accident conditions, including the LOCA-related design loads in and above the suppression pool. The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy, and energy from other sources such as the secondary system and metal-water reactions, including the recombination of hydrogen and oxygen. The evaluation of a containment functional design includes calculating the various effects associated with the postulated rupture in the primary or secondary coolant system piping.

A detailed description and definition of hydrodynamic loading conditions for the containment structure design is in Appendix 3B of the ABWR DCD.

6.2.1.2 *Summary of Application*

Section 6.2.1 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 6.2.1 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. Appendix 6C to the STP Units 3 and 4 COL FSAR also incorporates by reference Appendix C of the certified ABWR DCD, Revision 4. In addition, the applicant provides the following:

Tier 1 Departures

- STD DEP T1 2.4-2 Feedwater Line Break Mitigation

This departure adds differential pressure signals between the two feedwater lines to identify a feedwater line break (FWLB) and to then trip the condensate pumps.

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure replaces the reactor core isolation cooling (RCIC) turbine and pump system design with an integrated (monoblock) alternate turbine/pump system design.

- STD DEP T1 2.4-4 RHR, HPCF and RCIC Turbine/Pump NPSH

This change makes the ITAAC for the residual heat removal (RHR), high pressure core flooder (HPCF) and RCIC systems consistent with the STP Units 3 and 4 suction strainer design and the applicable regulatory guidance.

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This departure eliminates the requirements of the hydrogen control system to mitigate a design-basis LOCA hydrogen release.

Tier 2 Departures Requiring Prior NRC Approval

- STD DEP 6.2-2 Containment Analysis

This departure affects technical specifications (TS) and updates the ABWR DCD containment analysis in three areas:

- The modeling of flow and enthalpy into drywell for the feedwater following an FWLB
- The modeling of the DCVs for the FWLB and main steam line break (MSLB)
- The modeling of decay heat

- STD DEP 3B-2 Revised Pool Swell Analysis

This departure updates the hydrodynamic loads analysis to incorporate a new analytical method for the pool swell compared to the method described in the ABWR DCD. The applicant identifies this departure as requiring prior NRC approval. Because the applicant does not have access to the analytical codes described in ABWR DCD Section 3B, it was necessary to use an alternate method for performing the revised pool swell analysis.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 3B-1 — Equation Error in Containment Impact Load Description

This departure corrects the error in an equation that is used to calculate the impact load on flat structures inside the wetwell.

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

This departure incorporates the new complex ECCS (e.g., cassette type) strainer design.

- STD DEP Admin — Administrative Departure

The applicant provides editorial changes in FSAR Section 6.2.1; Subsections 6.2.1.1.7 and 6.2.4.3.2.1.2; Sections 3B.5 and 3B.7; and Subsections 3B.2.2.3, 3B.3.3, 3B.4.2.3, 3B.4.3.2.1, and 3B.4.3.3.3.3.

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.).

- STD DEP Vendor

In Appendix 6C, the applicant replaces “Toshiba” for “General Electric” or “GE,” which indicates a change in the reactor vendor.

COL License Information Items

- COL License Information Item 6.4 — Suppression Pool Cleanliness

In order to maintain the suppression pool cleanliness (in support of preventing ECCS suction strainer plugging in accordance with Subsection 6.2.1.7 and Appendix 6C), the applicant proposes maintenance inspections during outage periods for removing sediment and floating or sunk debris from the suppression pool that the suppression pool cleanup (SPCU) system does not already remove.

- COL License Information Item 6.5 — Wetwell-to-Drywell Vacuum Breaker Protection

The applicant proposes a vacuum breaker shield (consisting of a solid "V"-shaped plate) below each vacuum breaker to protect the vacuum breaker valves from LOCA pool swell loads.

6.2.1.3 *Regulatory Basis*

The regulatory basis for the information incorporated by reference is in NUREG–1503. In addition, the regulatory guidance for the containment functional design and the associated acceptance criteria is in Section 6.2.1 of NUREG–0800.

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

The review and acceptability of STD DEP 6.C-1 are based on compliance with 10 CFR 50.46(b)(5), as it relates to debris protection for ECCS strainers, and on conformance to the guidance in RG 1.82, Revision 3 and Topical Report NEDO-32686-A (“Utility Resolution Guidance for ECCS Suction Strainer Blockage”).

The review and acceptability of COL License Information Item 6.4 are based on following the guidance of RG 1.82 Revision 3, “Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident,” and NRC Bulletin No. 93-02, “Debris Plugging of Emergency Core Cooling Suction Strainers.” There is no regulatory guidance in NUREG–0800 for reviewing COL License Information Item 6.5 on the design of the vacuum breaker protection structure.

6.2.1.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.2.1 and Appendix 3B of the certified ABWR DCD. The staff reviewed Section 6.2.1 and Appendix 3B of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and the information incorporated by reference address the required information relating to the primary containment functional design.

The staff reviewed the following information in the COL FSAR:

Tier 1 Departures

The Tier 1 departures identified by the applicant in this chapter require prior NRC approval, and the full scope of the technical impact may be evaluated in other chapters of this SER accordingly. For more information, there is a listing of all FSAR sections affected by these Tier 1 departures in Part 7, Section 5.0 of the COL application. In addition, compliance with 10 CFR Part 52, Appendix A, Section VIII.A.4 for Tier 1 departures will be addressed by NRC staff in a future exemption evaluation.

- STD DEP T1 2.4-2 Feedwater Line Break Mitigation

This departure adds differential pressure signals between the two feedwater lines to identify an FWLB in the containment. Additionally, the departure also implements a condensation pump

[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.

trip, which is activated if the high drywell pressure signal exists in conjunction with the added differential pressure signals between the two feedwater lines.

This review focuses on the evaluation of the impact of this departure on the maximum containment pressure/temperature and the suppression pool hydrodynamic loads following a design-basis LOCA (FWLB) inside the containment, as it relates to this section of the FSAR. The FWLB is the limiting DBA for the ABWR containment. The staff found that implementing this departure will reduce the challenges to the containment pressure design value by limiting the release of mass and energy (M&E) to the containment following the FWLB. The departure is therefore acceptable.

This departure is also evaluated in SER Chapters 7, 14, 16, and 19.

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure changes the design of the RCIC turbine and pump assembly in favor of an improved design. The new RCIC turbine/pump is a monoblock design consisting of a horizontal, two-stage centrifugal water pump driven by a steam turbine contained in a turbine casing integral with the pump casing. This improved design simplifies the system due to (a) the monoblock design that places the pump and turbine within the same casing, (b) not requiring a shaft seal, (c) not requiring a barometric condenser, (d) not requiring an oil lubrication or oil cooling system because the system is totally water lubricated, (e) not requiring a steam bypass line for startup, (f) the simpler auxiliary subsystems, and (g) not requiring a vacuum pump and associated penetration piping or isolation valves.

The NRC staff's review focused on evaluating the effects of this departure on suppression pool hydrodynamic loads following a design-basis LOCA inside the containment. In COL FSAR Appendix 3B Subsection 3B.4.4.1, "Exhaust Steam Condensation Loading," the applicant states that the departure eliminates the requirement of ASME Code Class 2 piping for the steam turbine exhaust. Furthermore, the applicant incorporates a qualitative requirement that the turbine exhaust piping, including the RCIC sparger, will be designed to retain piping pressure integrity and functional capability. The staff does not expect the above changes to influence the condensation phenomenon that occurs at the exit of the RCIC steam turbine exhaust piping (inside the suppression pool) following a discharge of a steam/air mixture. Therefore, this departure will not affect the suppression pool hydrodynamic loading conditions following a design-basis LOCA. These loads are bounded by those resulting from the condensation oscillations (CO), chugging (CH), and SRV discharge. Within the review scope of Section 6.2, the staff found this departure acceptable.

This departure is also evaluated in SER Sections 6.3, 5.4.6, and 14.3.

- STD DEP T1 2.4-4 RHR, HPCF and RCIC Turbine/Pump NPSH

The NRC staff evaluated STD DEP T1 2.4-4 and found it accetable as discussed in this section under Departure STD DEP 6C-1 of Tier 2 Departures Not Requiring Prior NRC Approval.

- STD DEP T1 2.14-1 Elimination of Hydrogen Recombiner Requirements

In response to the modifications in 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors," which eliminate the requirements of hydrogen control systems to mitigate a design-basis LOCA hydrogen release, this departure eliminates the flammability control system (FCS). Also, the hydrogen and oxygen monitoring instruments are no longer classified as Category 1.

The NRC staff's review evaluated the effects of this departure on the design bases for the containment functional design. The FCS consists of two redundant hydrogen recombiners. Eliminating the FCS also eliminates the design-basis requirement that hydrogen recombiners establish and maintain the inert atmosphere in the containment during normal operating conditions. The updated design basis for the containment functional design states that the atmospheric control system (ACS) establishes and maintains the containment atmosphere to less than 3.5 percent by volume oxygen during normal operating conditions.

This departure incorporates changes to regulations that occurred after the issuance of the design certification for the ABWR. The design change satisfies 10 CFR 50.44(c) and is consistent with the guidance in RG 1.7, "Control of Combustible Gas Concentrations in Containment." Therefore, the staff found this departure acceptable.

Tier 2 Departures Requiring Prior NRC Approval

- STD DEP 6.2-2 Containment Analysis Methodology

This departure updates the ABWR DCD containment analysis. The DCD analysis was performed using the methodology developed by General Electric (GE) for the Mark III containment (NEDO-20533) (Ref. 6.2-1). The applicant identifies corrections in and improvements to the DCD analysis and documents the revised analysis in Technical Report WCAP-17058 (Ref. 6.2-2). The applicant uses the GOTHIC code (version 7.2a) for the revised containment analysis and incorporates the following corrections identified in the DCD analysis:

- The modeling of M&E released into drywell for the feedwater following an FWLB.
- The modeling of the DCVs for the FWLB and MSLB.
- The modeling of decay heat.

In addition, this departure includes the following changes:

- Updates the suppression pool temperature limit from the DCD-specified value of 97.2 °C to a value of 100 °C.
- Revises the assumed elapsed time between the start of the LOCA and the initiation of suppression pool cooling and containment sprays from 10 minutes to 30 minutes.

In the ABWR DCD containment analysis for the FWLB, the maximum possible feedwater flow rate was calculated to be 164 percent of the nuclear boiler rated (NBR) flow, which is based on the response of the feedwater pumps to an instantaneous loss of discharge pressure. Premised on the assumption that the feedwater control system would respond to the decreasing RPV water level by demanding increased feedwater flow, and because there was no FWLB mitigation logic in the certified ABWR design, this maximum feedwater flow was assumed by the applicant to continue for 120 seconds. Subsequently, an analysis performed by GE after the design certification determined that these assumptions were non-conservative (Ref. 6.2-3). Therefore, for the updated STP Units 3 and 4 containment analysis (for the FWLB accident), the applicant uses a revised feedwater M&E release rate derived from predicted characteristics of a typical feedwater system. The applicant states that the conservatism of the assumed M&E will be confirmed after completing the detailed condensate and feedwater designs. In **RAI 06.02.01.01.C-14,** the staff asked the applicant to clarify when the FWLB simulation will be available to the staff. **RAI 06.02.01.01.C-14** was tracked as **Open Item 06.02.01.01.C-14** in the SER with open items. In a response dated May 27, 2010 (ML101530167), the applicant commits to confirm the feedwater system side break M&E release rate used in the FWLB simulation, as part of the analyses to be performed as required by ITAAC (Inspections, Tests, Analyses, and Acceptance Criteria) Item No. 4 in ABWR DCD Tier 1, Table 2.14.1. The applicant also commits to update the departures report to clarify that this confirmation analysis will be performed as a part of the ITAAC. The applicant's response resolved **Open Item 06.02.01.01.C-14**. The staff verified that Revision 4 of the STP Units 3 and 4 COL departures report incorporates the appropriate updates, and **RAI 06.02.01.01.C-14** is now resolved**.**

The applicant also indicates that in order to provide additional assurance, a safety-related FWLB mitigation signal will be added to the STP Units 3 and 4 design, which will add safety-related instrumentation to detect the FWLB based on a high differential pressure between feedwater lines coincident with the high drywell pressure to trip the condensate pumps (see the evaluation of Departure STD DEP T1 2.4-2). However, this automated condensate pump trip is not credited and adds conservatism in the revised STP Units 3 and 4 containment analysis.

The applicant states that the ABWR DCD containment analysis does not properly simulate the horizontal vent portion of the vent system and incorrectly models the vent clearing time. The revised STP Units 3 and 4 containment analysis used the DCV loss coefficients and considered the horizontal vents. The calculated total effective vent loss coefficient included frictional losses at the DCV inlet (including losses due to the trash rack at the entrance to the DCVs), DCV exit losses, vertical vent pipe inlet losses, and horizontal vent turning losses. The applicant applied this total effective vent loss at the entrance of the vertical vent pipe. The staff issued **RAI 06.02.01.01.C-15** requesting the applicant to update the vent loss coefficient values shown in Table 6.2.1 of the FSAR accordingly. The staff tracked **RAI 06.02.01.01.C-15** as **Open Item 06.02.01.01.C-15** with the SER open items. The applicant's response dated May 27, 2010, states that Table 6.2-1 of the STP Units 3 and 4 COL FSAR will be revised to reflect the modified vent loss coefficients. This response resolved **Open Item 06.02.01.01.C-15.** The staff verified that the applicant has appropriately revised Table 6.2-1 in the STP Units 3 and 4 COL FSAR, Revision 4 and **RAI 06.02.01.01.C-15** is therefore resolved**.**

The applicant also states that the decay heat curves used in the DCD analysis are based on the best estimate of ANSI/American Nuclear Society (ANS)-5.1–1979 and are considered to be nonconservative for the long-term containment loading analysis. To address this nonconservatism, the decay heat curves used in the updated containment analysis were revised to reflect ANSI/ANS-5.1–1979, with 2-sigma uncertainty included.

The staff reviewed the updated STP Units 3 and 4 containment analysis documented in WCAP-17058 (Ref. 6.2-2) and performed independent confirmatory calculations using the

MELCOR 1.8.6 (version YT) computer code. The STP Units 3 and 4 containment analysis documented in WCAP-17058 includes discussions related to the implementation of a methodology consistent with the ABWR DCD/NEDO-20533 approach utilizing the GOTHIC code. WCAP-17058 also provides a detailed comparison of the ABWR DCD/NEDO-20533 containment analysis methodology with the GOTHIC-based approach and the revised STP Units 3 and 4 ABWR containment analyses using the GOTHIC model, by incorporating the corrections identified in the DCD analysis. The comparison of the transient results based on the GOTHIC calculations to those of the ABWR DCD for the short- and long-term FWLB and MSLB scenarios shows that the GOTHIC-calculated results are in agreement with those of the ABWR DCD. For this comparison, the GOTHIC analysis used input assumptions and M&E release boundary conditions to the containment similar to those used in the ABWR DCD analysis. This comparison confirmed the ability of the GOTHIC code to reproduce the ABWR DCD/NEDO-20533 results. Furthermore, the GOTHIC model was updated to correct the nonconservative assumptions in the DCD analysis described above. In the updated GOTHIC analysis, the vessel-side M&E releases for the FWLB and MSLB cases were calculated using the Westinghouse BWR LOCA M&E release calculation methodology, which is based on the Westinghouse GOBLIN code. WCAP-17058 (Ref. 6.2-2) shows that the total GOBLIN-calculated M&E releases for the FWLB and MSLB cases are higher compared to the ABWR DCD/NEDO-20533 results and are therefore more conservative.

The applicant's GOTHIC-calculated peak drywell pressure and temperatures are higher than those reported in the ABWR DCD. The GOTHIC-calculated drywell peak pressure for the short-term FWLB is 281.8 kPaG. The design pressure for the drywell is 309.9 kPaG, which provides a margin of approximately 10 percent above the calculated peak pressure in the drywell. This GOTHIC-calculated drywell pressure margin is lower than the margin estimated in the ABWR DCD (i.e., 15 percent). The GOTHIC-calculated maximum drywell atmospheric temperature for the MSLB is 173.2 °C, which exceeds the drywell design temperature of 171.1 °C (by 2.1 °C) for about 2 seconds. However, due to thermal inertia, components in the drywell structures (in particular, the upper head seals) will not have sufficient time to reach the design limit temperature during such a short amount of time. The GOTHIC-calculated wetwell peak pressure (217.2 kPaG) (occurring in the long-term MSLB) and maximum gas space temperature (98.6 °C) (occurring in the long-term MSLB) are lower than the wetwell design pressure (309.9 kPaG) and the design temperature (104 °C). The staff issued **RAI 06.02.01.01.C-18** requesting the applicant to explain the discrepancy between the analysis discussed above and the values provided in COL FSAR Revision 3. The staff tracked this RAI as **Open Item 06.02.01.01.C-18** in the SER with open items. The applicant's response to this open item dated May 27, 2010, states that Table 6.2-1 of the STP Units 3 and 4 COL FSAR will be revised to reflect the updated values. This response resolved **Open Item 06.02.01.01.C-18.** The staff verified that the applicant has appropriately revised Table 6.2-1 in the STP Units 3 and 4 COL FSAR Revision 4, and **RAI 06.02.01.01.C-18** is now resolved**.**

The GOTHIC-calculated maximum suppression pool temperature (99.5 °C) exceeded the DCD-specified suppression pool temperature limit of 97.2 °C. However, the applicant increased the design suppression pool temperature limit to 100 °C and has revised the ECCS pump net positive suction head (NPSH) calculations accordingly by assuming the updated suppression pool design temperature limit. The staff found this approach acceptable.

The GOTHIC-calculated drywell-to-wetwell peak differential pressure is 148 kPaD, which provides a design margin of 18 percent when compared to the design drywell-to-wetwell differential pressure of 172.6 kPaD. The staff issued **RAI 06.02.01.01.C-16** requesting the applicant to update the FSAR accordingly. The staff tracked this issue as **Open Item 06.02.01.01.C-16** in the SER with open items. The applicant's response dated May 27, 2010, states that Table 6.2-1 of the STP Units 3 and 4 COL FSAR, will be revised to reflect the modified value of the maximum drywell-to-wetwell differential pressure. This response resolved **Open Item 06.02.01.01.C-16**. The staff verified that the applicant has appropriately revised Table 6.2-1 in the STP Units 3 and 4 COL FSAR Revision 4, and **RAI 06.02.01.01.C-16** is now resolved**.**

The NRC staff's confirmatory calculations include base-case simulations and several sensitivity cases. The base-case simulations use one drywell node containment model (similar to the GOTHIC model) with GOTHIC/GOBLIN-calculated M&E releases as boundary conditions. Other MELCOR model assumptions for the base case are similar to the GOTHIC model. The sensitivity case simulations include a study of sensitivity-to-drywell nodalization, an effective vent loss coefficient, the drywell node volume (only for a short-term FWLB simulation), and MELCOR-calculated M&E releases. The results of the confirmatory calculations show that the MELCOR base-case predictions agree with the GOTHIC predictions, except for the case of a short-term FWLB. The MELCOR-predicted peak drywell pressure is 287.7 kPaG, which provides a slightly reduced drywell pressure margin (i.e., 8 percent) compared to the GOTHIC prediction (i.e., 10 percent). Furthermore, the results of the MELCOR sensitivity calculations show that except for the drywell volume assumption used in the FWLB simulation, all other GOTHIC model assumptions studied in the sensitivity analysis were determined to be conservative. The ABWR DCD and the STP Units 3 and 4 containment analyses of the short-term FWLB scenario credited only 50 percent of the actual volume of the lower drywell. The MELCOR short-term FWLB sensitivity calculation with 100 percent lower drywell volume showed a peak drywell pressure of 298.7 kPaG, which resulted in a reduced drywell pressure margin (i.e., ~4 percent).

In addition, the confirmatory analyses found/identified that the applicant needed to provide additional information regarding the following issues:

- For the short-term MSLB accident scenario, the GOBLIN-calculated total break flow rate is lower than that of the ABWR DCD during the initial time interval of 0 to 3 seconds following accident initiation. The NRC confirmatory MELCOR calculation of the break flow rate for the MSLB accident scenario also shows results similar to those of the ABWR DCD. Because the maximum drywell temperature occurs in the short-term MSLB simulation (at around 4 seconds following the accident), the observed differences in the break flow rate estimate may affect the calculated maximum drywell temperature. The reason for the discrepancy in the break flow rate calculations is unclear. Therefore, the staff issued **RAI 06.02.01.01.C-9** requesting the applicant to provide additional information on the volumes of various pipe sections in the main steam line and the various pressure losses inside the main steam line system. The applicant's response to the RAI dated January 20, 2010 (ML100220492), states that the differences between the GOBLIN-calculated break flow rate and the ABWR DCD break flow rate are due to the additional main steam line piping detail present in the GOBLIN model. On the basis of the detailed main steam line design data that the applicant includes in the RAI response dated January 20, 2010, the staff updated the MECLOR model and revised the confirmatory calculation. The calculation results show that the MECLOR-calculated break flow rate is close to the GOBLIN-calculated break flow rate. This finding resolved **RAI 06.02.01.01.C-9**.
- The results of the NRC confirmatory MELCOR base-case calculation for the long-term MSLB will be updated upon receiving information on GOTHIC/ GOBLIN ECCS mass

flow beyond 600 seconds. The staff issued **RAI 06.02.01.01.C-10** requesting the applicant to provide this information. The applicant provided the requested information in a response dated January 20, 2010. The staff updated the results of the MELCOR confirmatory base-case calculation for the long-term MSLB. This information resolved **RAI 06.02.01.01.C-10**.

- The STP Units 3 and 4 containment analysis in WCAP-17058 does not document the results of the long-term FWLB accident simulation. Therefore, the staff issued **RAI 06.02.01.01.C-11** asking the applicant for the results of the containment pressure/temperature analysis and the M&E release rate to the containment for the long-term FWLB accident simulation. The applicant's response to this RAI dated January 21, 2010 (ML100250134), provides the requested information. The staff found the response acceptable, and **RAI 06.02.01.01.C-11** is resolved. Chapter 16 of this SER addresses the effect of this departure on the TS.

- STD DEP 3B-2 Revised Pool Swell Analysis

The departure updates the ABWR DCD pool swell analysis. During a postulated LOCA inside the drywell, the wetwell region will be subjected to the sequential hydrodynamic loading conditions due to pool swell, COs, and CH. Following the LOCA and after the water is cleared from the vents, the air/steam mixture from the drywell flows into the suppression pool and creates a large bubble at the vent exit as it exits into the pool. At the vent exit, the bubble expands to the suppression pool hydrostatic pressure as the air/steam mixture flow continues from the pressurized drywell. The water ligament above the expanding bubble is accelerated upward, which gives rise to a pool swell phenomena that typically lasts for a couple of seconds. During this pool swell phase, the wetwell region is subjected to

- loads on the suppression pool boundary and drag loads on structures initially submerged in the pool
- loads on the wetwell gas space
- the impact of drag loads on structures above the (initial pre-accident) pool surface

The DCD pool swell analysis was performed to determine the maximum pool surface elevation, the peak pool surface velocity, the peak wetwell gas space pressure, and the peak bubble pressure (before the bubble breakthrough of the pool surface) following a design-basis LOCA. These pool swell parameters were later used to estimate the pool swell hydrodynamic loads listed above. The DCD pool swell analysis was performed using GE's PICSM computer code, which was validated against the MARK III pressure suppression test facility (PSTF) data applied to the ABWR pool swell analysis. In NUREG–1503, NRC staff accepted the submitted pool swell analysis based on the relevant test data rather than on the approval of the PICSM computer code.

The applicant is required to update the pool swell analysis in order to address the effects of the changes in the containment pressure response for LOCA events, as described in STD DEP 6.2-2. However, the applicant no longer has access to the PICSM computer code. Therefore, the applicant has proposed an alternate method for performing the revised pool swell analysis that uses a calculation approach similar to that used in the DCD, with different assumptions and analytical software. The applicant has benchmarked the proposed alternate method by comparing it against one MARK III PSTF test (test 1 from PSTF series 5806). The staff verified the applicant's benchmarking analysis through the NRC-conducted audit (ML092790335), and the applicant has documented the revised ABWR pool swell analysis in Technical Report UTLR-0005 (Ref. 6.2-4).

NRC staff reviewed the updated STP Units 3 and 4 pool swell analysis. The applicant's alternate pool swell analysis method uses the GOTHIC code. Similar to the DCD model approach, the drywell pressure is applied as a boundary condition in the GOTHIC model. The applicant documents the results from comparing the GOTHIC model with the ABWR DCD pool swell analysis and presents the revised ABWR pool swell analysis in Technical Report UTLR-0005 (Ref. 6.2-4). For the comparison against the DCD methodology, the GOTHIC model used the same drywell pressure boundary condition that the DCD analysis used. For the revised ABWR pool swell analysis using the GOTHIC model, the drywell pressure transient was calculated using the GOTHIC/GOBLIN code-based methodology described in Departure STD DEP 6.2-2 (this departure accounts for the corrections and improvements in the DCD containment analysis).

The applicant's comparison of the GOTHIC model predictions against one of the PSTF experiments and the DCD model shows that the GOTHIC model provides bounding estimates of pool swell parameters when compared to the experimental data and the DCD model predictions of pool swell parameters (except for the maximum gas space pressure, which is slightly lower than the value reported in the DCD). Furthermore, as shown below, the values of pool swell load parameters predicted in the revised ABWR pool swell analysis using the GOTHIC model are higher compared to the ABWR DCD values:

- Maximum pool surface elevation 8.8 m (DCD analysis, 7.0 m).
- Maximum pool surface velocity 10.9 m/second (s) (DCD analysis, 6.0 m/s).*
- Maximum wetwell gas space pressure 146 kPaG (DCD analysis, 108 kPaG).
- Maximum bubble pressure 195 kPaG (DCD analysis, 133 kPaG).

The NRC confirmatory STP Units 3 and 4 pool swell analyses include the base-case calculation (with best estimate model input parameters) and several sensitivity case calculations to study the impact of the key modeling assumptions governing the parameters. The analyses show that the best estimate pool swell parameters predicted by the NRC model are much lower when compared to the STP predictions using the GOTHIC model. This finding confirms that the GOTHIC model input assumptions are conservative.

The NRC confirmatory analysis indicates that the GOTHIC model assumption for the vent loss coefficient is more conservative when compared to the DCD model input assumptions. Also, for the comparison against the experimental data, the GOTHIC model uses this conservative input assumption and shows that the model predictions bound the experimental data. In order to establish confidence that the GOTHIC model results are conservative, the staff issued **RAI 06.02.01.01.C-12** requesting the applicant to perform additional experimental benchmark calculations. Consequently, the applicant compared the GOTHIC model predictions against two additional MARK III PSTF tests (i.e., test 2 from PSTF series 5806 and test 10 from PSTF series 5801). These results show that the GOTHIC model conservatively bounds both the pool swell and the pool surface velocity measured in the experiments. The staff found these findings acceptable, and **RAI 06.02.01.01.C-12** is therefore resolved.

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* With multiplier 1.1.

Also, the staff issued **RAI 06.02.01.01.C-13** requesting the applicant to perform a sensitivity study to assess the impact of spatial nodalization on the GOTHIC-predicted loads. The applicant performed the requested sensitivity calculations and summarized the results in a response dated January 20, 2010 (ML100220492). The applicant's analysis showed that the pool swell and swell velocity are not very sensitive to node size. These results resolved **RAI 06.02.01.01.C-13**.

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

- STD DEP 3B-1 Equation Error in Containment Impact Load Description

The applicant corrects a multiplying factor in an equation used to calculate pulse duration for a flat target. As the pool level rises during the pool swell, structures or components located above the initial pool surface (but lower than its maximum elevation) will be subjected to the water impact load. The calculation of pulse duration is necessary for estimating the pool swell impact loads. ABWR DCD Appendix 3B, Subsection 3B.4.2.3 provides two equations for calculating pulse duration for a flat target. The equation for impact velocity (V) < 2.13 m/s is

$$T = (0.0016 \times W)$$

Where:

T is the duration of impact (seconds), and
W is the width of the flat structure (meters).

The multiplying factor for W is incorrect because its dimensions are seconds per foot instead of seconds per meter, as required in this case. This departure corrects the multiplying factor from 0.0016 s/ft to 0.0052 s/m.

The applicant states that this change does not affect the design or function of the structures and components that can be impacted by a suppression pool swell. The applicant also states that the correct loads were used for the structural analyses to show that the structures and components are able to withstand the loads adequately, without any failure.

The staff confirmed that this change involves a correction in a formula but does not alter the hydrodynamic method of evaluating the pool swell load effect. This departure does not alter any method used in the design bases or safety analyses.

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

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

The STP Departures Report states that a departure from Appendix 6C incorporates the new complex ECCS strainers (e.g., cassette-type strainer) design per NUREG/CR–6224, NUREG/CR–6808, and the “Utility Resolution Guidance for ECCS Suction Strainer Blockage,” NEDO-32868-A. The ECCS strainer design also affects the description and the available NPSH of the ECCS pumps. The report introduces additional mitigating features such as the use of reflective metallic insulation (RMI) for large bore piping, the Inservice Inspection Program as a surveillance requirement, temporary filters during post-construction system testing, and a Foreign Material Exclusion Program.

The applicant's evaluation in accordance with 10 CFR Part 52, Appendix A, Section VIII, Item B.5 determined that these departures do 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. In addition, the applicant's process for evaluating departures and changes to the DCD is subject to NRC inspections.

Although the applicant identifies Tier 2 Departure STD DEP 6C-1 as not requiring prior NRC approval, the staff found it necessary to evaluate this departure within the scope of conformance with RG 1.82, Revision 3, because the applicant has committed to conform to this guidance. The staff thus found it necessary to determine whether the applicant conforms to this guidance.

In STP Units 3 and 4 COL FSAR, Revision 2, Appendix 6C states that the ABWR design commits to the guidance in RG 1.82 and the "Utility Resolution Guidance for ECCS Suction Strainer Blockage," NEDO–32686-A. However, the information in the application was not sufficient for the staff to confirm that STP Units 3 and 4 ECCS debris strainers conform to the above guidance. Therefore, NRC staff issued **RAI 06.02.02-1** requesting the applicant to submit a calculation report on sizing the suppression pool recirculation suction debris strainers. The staff reviewed the report to determine whether the strainers meet the guidance in RG 1.82, Revision 3. In a letter dated April 29, 2009 (ML091270491), the applicant states the intention to use the same ECCS suction debris strainers design that a referenced Japanese ABWR (RJABWR) uses. The applicant also states that a summary calculation report on that design will be available for the staff to audit. During a staff audit on June 30 and July 1, 2009 (ML092370709), the staff reviewed that summary calculation report and two additional documents on the ECCS suction debris strainers design. During a review of the applicant's response and the audit, the staff identified several issues in need of additional information for the staff to complete the review. The staff issued **RAI 06.02.02-6**, which the applicant responded to in letters dated September 28 and October 29, 2009, and February 15, 2010. The following information is a summary of the staff's requests and the applicant's responses.

The documents that the staff reviewed during the audit did not account for miscellaneous debris (equipment tags, tape, and stickers or placards affixed by adhesives) that the staff had considered during the resolution of the Generic Safety Issue (GSI) 191 Program on the effect of debris accumulation on ECCS suction strainers of pressurized water reactors (PWRs). The staff issued **RAI 06.02.02-6** requesting the applicant to account for the miscellaneous debris. In a letter dated September 28, 2009 (ML092730448), the applicant's response states that based on operating experience at STP Units 1 and 2 (operating the PWR), each strainer will be assumed to have the openings of two cassettes blocked by miscellaneous latent debris (e.g., small pieces of plastic, tape, sheets of paper, health physics low-dose sign). Considering that following a LOCA, two RHR pumps and one HPCF pump would be in operation and each pump has two strainers, the total blocked area of the strainers would be equal to 0.74 m^2 (8 ft^2). The staff noted that STP Units 1 and 2 have assumed a significantly larger blocked area for miscellaneous latent debris (7.0 m^2 [75 ft^2]) that would be transported to the sump (GSI-191 Program, Generic Letter [GL] 2004-02 Second Response, dated August 31, 2005). Therefore, the staff issued **RAI 06.02.02-26** requesting the applicant to justify the basis for assuming that two cassettes per strainer will be blocked by miscellaneous latent debris.

The applicant's response to RAI 06.02.02-26 in a letter dated April 14, 2010 (ML101090142), agrees to update STP Units 3 and 4 FSAR Subsection 6C.3.1.2 to state that "[w]ith regard to LOCA-generated miscellaneous debris, the design of STP 3&4 minimizes the potential for such debris by specifying secure restraints, such as high tensile strength aircraft cable or specially designed bands, to secure equipment ID tags onto components located inside containment." This response addresses the staff's concern and is therefore acceptable. The staff verified that the applicant has appropriately updated Revision 4 of STP Units 3 and 4 COL FSAR Subsection 6C.3.1.2, and this RAI is now resolved.

During the audit, the applicant stated that subsequent to the response to **RAI 06.02.02-1**, the applicant had decided to eliminate all fiber insulation from the STP Units 3 and 4 primary containments. The staff was concerned that not accounting for any fiber debris in the design of debris strainers would become an issue if fiber is found in the ABWR containment during operation. The staff issued **RAI 06.02.02-6** requesting the applicant to either account for the possibility of having some fiber in the containment in terms of latent debris or to confirm with a Foreign Material Exclusion Program the elimination of all fiber from the STP Units 3 and 4 primary containments. In a letter dated September 28, 2009, the applicant's response states that no fibrous insulation is allowed in the primary containment. In addition, for operational flexibility, 0.03 m^3 (~ 1 ft^3) of latent debris will be assumed in the strainer head loss calculation. The staff determined that this approach is reasonable and addresses the staff's concern. Although the applicant committed to eliminate all fiber insulation from the containment, the applicant's analysis assumes one cubic foot of fiber and is thus conservative.

During the audit (ML092370709), the applicant stated that the thermal insulation in the STP Units 3 and 4 primary containments would be all stainless steel RMI. The staff was concerned that the applicant may not be able to use RMI for some small bore piping because of their locations. The applicant may thus have to use small quantities of other types of insulation, such as calcium silicate and fiber. Therefore, the staff issued **RAI 06.02.02-6** requesting the applicant to account for that possibility in the debris strainer design. In a letter dated September 28, 2009, the applicant's response states that STP Units 3 and 4 were designed using state-of-the-art, 3-dimensional (3-D) computer-aided design/drafting tools. So all piping arrangements, including small-bore piping, were designed to account for the needed clearance for thermal insulation, and there was no need to account for non-RMI thermal insulation in the STP Units 3 and 4 primary containments. The staff determined that this response addresses the staff's concern.

The applicant's response to **RAI 06.02.02-1** states that the latent debris defined in the Utility Resolution Guidance (URG), which was used for the RJABWR testing, is considered bounding for STP Units 3 and 4. The URG on latent debris includes paint chips, rust flakes, sludge, and dust. However, the URG-proposed generic values are based on the operating experience with BWRs, and the ABWR is a newer plant whose operating experience was not considered in determining the URG-proposed values. The staff issued **RAI 06.02.02-6** requesting the applicant to confirm that the quantities of latent debris used in the design are consistent with the operating experience of ABWRs or to propose a plan that will confirm those values later. The applicant responded to this RAI in a letter dated September 28, 2009. As discussed later in this section in the paragraph related to protective coatings, the staff evaluated the quantity of paint chips assumed in the applicant's analysis and found this assumption acceptable.

According to the response to RAI **06.02.02-6**, the applicant was going to confirm that quantities of rust flakes and sludge are conservative based on operational information from the Toshiba Electric Power Company (TEPCO) on quantities of material obtained from the SPCU systems at Kashiwazaki-Kariwa Units 6 and 7, which are the oldest operating ABWRs. The staff reviewed these data and determined that the assumed quantities of rust flakes and sludge for the STP Units 3 and 4 ECCS suction debris strainer design are greater than the quantities from TEPCO and are thus conservative.

The applicant also states that the URG estimate of the quantity of dust is conservative. As discussed below, the staff determined that the Suppression Pool Cleanliness Program that will be in effect will limit the quantity of dust that will be present in the STP Units 3 and 4 containments. Based on this program, the staff determined that the URG estimate of the quantity of dust that was used for the STP Units 3 and 4 ECCS suction debris strainer was conservative.

The applicant's supplemental response to **RAI 06.02.02-6** dated October 29, 2009 (ML093090336), proposes a change to the STP Units 3 and 4 FSAR that "The ECCS suction strainer design to be used on STP 3&4 is the same as the design for the Reference Japanese ABWR." The staff determined that the information in the STP Units 3 and 4 FSAR would not provide sufficient details about the STP Units 3 and 4 ECCS suction debris strainer design. The staff issued **RAI 06.02.02-20** requesting the applicant to provide sufficient information in the FSAR on the strainers, rather than referring to a plant for which information is not readily available. In a letter dated January 13, 2010 (ML100141735), the applicant's response to **RAI 06.02.02-20** provides details on the STP Units 3 and 4 ECCS suction debris strainer design as an update to the FSAR. In this proposed FSAR update, the applicant states that the ECCS suction strainers design to be used at STP Units 3 and 4 is the same design used for the RJABWR, and the STP Units 3 and 4 strainers will be at least as large as those for the RJABWR. The applicant also lists reasons why the STP Units 3 and 4 strainer design is conservative. The staff noted that the applicant did not list a nonconservatism. In this case, the nonconservatism is that the RJABWR strainer design is for the pump design flow rate, while the STP Units 3 and 4 strainer design is for the pump runout flow rate. As described later in this report, the applicant evaluated the effects of conservative and nonconservative assumptions. The staff found that the STP Units 3 and 4 strainer design is acceptable because it conforms to the guidance in RG 1.82, Revision 3, regarding the NPSH. The staff requested the applicant to update the FSAR to include the nonconservatism. The staff found the applicant's response to this RAI acceptable and tracked it as **Confirmatory Item 06.02.02-20** for purposes of verification. The staff verified that the applicant has appropriately updated Revision 4 of the STP 3 and 4 COL with this information, and **Confirmatory Item 06.02.02-20** is now resolved**.**

It was not clear to the staff from applicant's submittals how the STP Units 3 and 4 ECCS suction debris strainers meet the regulatory positions outlined in RG 1.82, Revision 3. The staff issued **RAI 06.02.02-6** requesting the applicant to provide a table listing how the STP Units 3 and 4 ECCS suction debris strainer meets each regulatory position for BWRs that is stated in RG 1.82 Revision 3, or justify an alternate approach. In a supplemental response dated October 29, 2009 (ML093090336), the applicant provides the requested table. The staff noted that conformance to Regulatory Position 2.1.6, "Inservice Inspection," was missing in the response. The staff issued **RAI 06.02.02-24** requesting the applicant to provide the missing information. In a letter dated January 13, 2010 (ML100141735), the applicant's response to **RAI 06.02.02-24** provides information relating to compliance with Regulatory Position 2.1.6.

The staff determined that the information is acceptable, and **RAI 06.02.02-24** is closed. The remaining issues addressed in **RAI 06.02.02-6** are discussed below.

Following the audit (ML092670380), the staff determined that a detailed report on the design of the STP Units 3 and 4 ECCS suction debris strainers in accordance with RG 1.82 Revision 3 is required to complete the staff's review. The staff issued **RAI 06.02.02-6** requesting the applicant to submit a detailed calculation report on the design of the STP Units 3 and 4 ECCS suction debris strainers. In a letter responding to **RAI 06.02.02-6** dated October 29, 2009 (ML093090336), and a letter responding to RAI 06.02.02-14 Supplement 2 dated February 15, 2010 (ML100541568), the applicant submits three proprietary documents (Ref. 6.2-5 through 6.2-7) and summarizes relative conservatisms and nonconservatisms in the STP Units 3 and 4 design compared to those in the RJABWR design. Relative conservatisms identified by the applicant include the following:

- The RJABWR contains fiber insulation on small-bore piping in the zone of influence, which is transported to the suppression pool. The STP Units 3 and 4 design does not allow fiber insulation in the containment, and fiber on the small bore piping will be replaced with RMI. Head loss on the debris strainer due to RMI without fiber is negligible. The only fiber postulated to be in the STP Units 3 and 4 containment will be 0.03 m^3 (~ 1 ft^3) of latent fiber debris that will be assumed in the head loss calculation for operational flexibility.

- The RJABWR contains calcium silicate insulation that is transported to the suppression pool. Only RMI is allowed for thermal insulation inside the STP Units 3 and 4 primary containment. Calcium silicate is a significant contributor to head loss and STP Units 3 and 4 will eliminate this contributor to head loss.

- The RJABWR ECCS suction debris strainer design did not account for miscellaneous latent debris (equipment tags, tape, and stickers or placards affixed by adhesives), because the URG guidance that the RJABWR used does not have a requirement to account for such debris. The STP Units 3 and 4 ECCS suction debris strainer design does account for miscellaneous latent debris.

A relative nonconservatism is the assumption of a pump design flow rate in the RJABWR ECCS suction debris strainer design, compared with an assumption of a pump runout flow rate in the STP Units 3 and 4 design, which will result in a higher head loss. In STP Document U7-RHR-M-RPT-DESN-0003, Revision A (ML093130317), (Ref. 6.2-7), the applicant evaluates the impact of this nonconservative change and the above conservative changes on the STP Units 3 and 4 ECCS debris strainer design. This evaluation assumes only a partial replacement of fiber insulation in the STP Units 3 and 4 containments with RMI— 0.73 m^3 (25.8 ft^3) of fiber debris would be transported to the strainers. After performing the calculation in STP Document U7-RHR-M-RPT-DESN-0003, Revision A, the applicant decided to replace all of the fiber insulation in the STP Units 3 and 4 containments with RMI. The amount of fiber assumed for the calculation was reduced to 0.03 m^3 (~ 1 ft^3) of the latent fiber debris transported to the strainers. The evaluation in STP Document U7-RHR-M-RPT-DESN-0003, Revision A, shows that NPSH margins exist for both the RHR and HPCF pumps. After reviewing STP Document U7-RHR-M-RPT-DESN-0003, Revision A, the staff determined that based on the conservative and nonconservative changes cited above, NPSH margins exist for the RHR and HPCF pumps.

As discussed above, the staff found that applicant's responses to **RAI 06.02.02-6** and to the follow up RAIs are acceptable, and **RAI 06.02.02-6** is therefore closed.

Section 6C.2 of the STP Units 3 and 4 FSAR states:

> [t]he ABWR design also has additional features not utilized in earlier designs that could be used in the highly improbable event that all suppression pool suction strainers were to become plugged. The alternate AC (Alternating Current) independent water addition (ACIWA) mode of RHR allows water from the Fire Protection System to be pumped to the vessel and sprayed in the wetwell and drywell from diverse water sources to maintain cooling of the fuel and containment.

If the above feature is used in the long term, the containment would pressurize from a decrease in free volume as a result of a continuous addition of water into the containment. Therefore, the staff issued **RAI 06.02.02-3** requesting the applicant to explain and account for such pressurization. In a letter dated September 28, 2009 (ML092730448), the applicant's response states that the ACIWA mode of RHR for reactor vessel injection and drywell spray is analyzed in Section 19E.2.2 of the ABWR DCD. The operator actions associated with controlling the reactor pressure vessel and primary containment level control and injection from sources external to the primary containment (e.g., the ACIWA system) are included in the emergency procedure guidelines (EPGs) in FSAR Appendix 18A. These operator actions in the EPGs include precautions to maintain the primary containment water level and pressure low enough to preclude a primary containment failure and to terminate injection when required. The staff found the applicant's response acceptable because the operator's actions are included in the EPGs and conform to Regulatory Position 2.2 of R.G 1.82, Revision 3. Therefore, **RAI 06.02.02-3** is closed.

The ITAAC for the RHR, HPCF and RCIC systems in ABWR DCD Tier 1 Tables 2.4-1, 2.4-2, and 2.4-4 refer to a criterion of 50 percent blockage of pump suction strainers in determining the NPSH margin, as stated in RG 1.82 Revision 1. However, the applicant has committed to design STP Units 3 and 4 to conform with RG 1.82 Revision 3, which does not refer to the criterion of a 50 percent blockage of pump suction strainers. The staff issued **RAI 06.02.02-22** requesting the applicant to change the STP Units 3 and 4 FSAR to reflect conformance with the guidance in RG 1.82, Revision 3. In a letter dated January 13, 2010 (ML100141735), the applicant's response to **RAI 06.02.02-22** agrees to revise the FSAR and change the "50% blocked strainer" criterion to "analytically derived values for blockage of pump suction strainers based upon the as-built system" and to provide FSAR updates. The staff determined that the applicant's response addresses the staff's concern and is therefore acceptable. In Revision 4 of the STP Units 3 and 4 COL application Part 2, Tier 1 Section 2.4, the applicant adds Departure STD DEP T1 2.4-4, which revises the ITAAC for the RHR, HPCF, and RCIC systems as requested in RAI 06.02.02-22. The staff determined that Departure STD DEP T1 2.4-4 is consistent with the applicant's commitment to design STP Units 3 and 4 to conform with RG 1.82 Revision 3, and therefore, acceptable. **RAI 06.02.02-22** is closed.

Detailed evaluation of the ECCS strainer for structural integrity is in SER Section 3.9.3.

Protective coatings (i.e., paints) are a potential source of LOCA-generated debris in the containment. Such debris could potentially contribute to the plugging of ECCS suction strainers, downstream components, and fuel. The amount and size of the debris depend on the type, location, and condition of the coating. The potential for such debris to degrade emergency core cooling is discussed in NRC documents including GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss of Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment." As stated in COL FSAR Section 6C.1, the ABWR commits to following the guidance related to ECCS blockage in RG 1.82 and in Topical Report NEDO-32686-A, "Utility Resolution Guidance for ECCS Suction Strainer Blockage," (the URG).

The staff reviewed the applicant's information on protective coatings debris using the guidance in RG 1.82 Revision 3, Section 2.3.1.4 and the staff's guidance on using NEDO-32686-A. The staff's guidance on the URG is documented in an SER dated August 20, 1998 (Ref. 6.2-8). In addition, the staff used the coatings evaluation guidance issued in March 2008 for the resolution of GL 2004-02 regarding potential debris blockage of the PWR emergency recirculation (Ref. 6.2-9). This document supplements RG 1.82 and the URG on several topics, including coatings debris particle size. Conforming to this guidance provides one acceptable way to meet the requirements of 10 CFR 50.46 and to address the concerns expressed in GL 2004-02 related to coatings debris.

The applicant's response to **RAI 06.02.02-1,** on how STP Units 3 and 4 would conform to the guidance in RG 1.82 Revision 3 dated April 29, 2009 (ML091270491), indicates an assumed coatings debris quantity of 85 pounds based on the URG guidance for inorganic zinc with epoxy topcoat. This quantity, appropriately scaled, was used in the head loss testing for the RJABWR.

However, the applicant does not state whether the URG is conservative for STP Units 3 and 4 or describe a size distribution for the coatings. The need to determine coatings debris size for BWRs is listed in Regulatory Position 2.3.1.4 of RG 1.82. For evaluating suction strainers, the URG assumes that epoxy coatings are in chip form. The staff issued RAI 06.02.02-8 requesting the applicant to provide additional information about the determination of the amount of coatings debris and the particle size distribution.

The applicant's response to RAI 06.02.02-8 dated September 28, 2009 (ML092730448), clarifies the basis for assuming the 85 pounds coatings debris. In addition to being approved by the NRC for BWRs in the staff's SER on the URG, the applicant states that the assumed quantity is conservative for STP Units 3 and 4 because even though the 85 pounds applies to epoxy/inorganic coating systems, the ABWR will use only epoxy. Because the potential for blockage increases with the amount of debris, the staff determined that 85 pounds is more conservative than the 71-pound value in the URG guidance for epoxy-only debris, which is the type of coating system specified in the ABWR DCD. Therefore, the staff found this coatings debris quantity acceptable because the applicant's analysis conforms to the URG for the coating quantity.

With respect to the particle size, the head loss testing for the ECCS screens assumes 85 pounds of coatings debris in chip form, which is acceptable because it is consistent with the assumptions in the URG and the small amount of fiber. For downstream effects, the applicant provides more information in response to RAI 04.04-3 dated February 22, 2010 (ML100560113), regarding downstream effects testing for fuel assemblies. In that response, the applicant states that the coatings debris is assumed to entirely consist of fine particles. This approach assumes that all coatings debris will pass through the ECCS strainers and will reach the fuel assemblies. Therefore, the particles may be trapped on a fiber bed at the fuel assembly. The staff's March 2008 guidance (ML080230462) (Ref. 6.2-9) states that where there is a possibility of forming a thin fiber bed, coatings debris should be treated as fine particles that can be trapped by the bed (and contribute to head loss). Because fiber will be included in the fuel assembly testing for STP Units 3 and 4, but it is not yet known whether a thin fiber bed will form, the applicant's assumption of coatings debris as a fine particulate conforms to the staff's guidance and is conservative for fuel assembly testing.

The applicant's supplemental response to RAI 04.04-3 includes a proposed revision to FSAR Subsection 6C.3.1.9.3, which states that the coatings debris for downstream fuel effects testing will consist of small particles that are assumed to pass through the ECCS strainers. This testing is required by License Condition 06.02-1. The proposed FSAR revision includes a table that lists 38 pounds as the coatings debris quantity that will be scaled down for fuel assembly testing. The table includes a note explaining that this value is based on subtracting the 47 pounds of inorganic zinc (IOZ) coating from the 85 total pounds of coatings. This approach is consistent with the chemical effects analysis discussed below, which concludes that at the limiting pH, all of the IOZ will be dissolved and converted to chemical debris. The staff found this acceptable because it is consistent with the chemical effects analysis. The staff also noted that the test plan includes a large quantity of particulate debris from sources other than coatings. The staff found the particle size distribution acceptable because the applicant's approach of assuming that the coatings debris distribution will consist entirely of fine particles conforms to the staff's guidance on GL 2004-02 (Ref. 6.2-9) that supplements RG 1.82 and the URG, as described above. These proposed changes to the FSAR are being tracked as **Confirmatory Item 04.04-3**.

Therefore, for the reasons described above, the staff determined that the applicant's treatment of the coatings debris is acceptable, including the coatings debris quantity and assumed particle size, because they conform to RG 1.82, Revision 3 and the URG, as well as the supplemental guidance on coating particle size.

The term "chemical effects" refers to the possibility that interactions of materials in the containment environment will generate chemical precipitate debris that may contribute to blockage and head loss. In RG 1.82 Revision 3, Subsection 2.3.1.8 states that debris created from the thermal and chemical conditions in the containment pool should be considered in evaluations of long-term recirculation capability. NRC staff published detailed guidance on preparing response to GL 2004-02 in March 2008 for PWR licensees to evaluate plant-specific chemical effects (ML080380214) (Ref. 6.2-10). This includes guidance on using WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," (ML081150383) (Ref. 6.2-11). Separately, the staff issued an SER to approve, with limitations and conditions, the use of WCAP-16530-NP-A to evaluate chemical effects in PWR post-LOCA containment fluids. Conforming to this guidance provides one acceptable way to meet the requirements of 10 CFR 50.46, as they relate to chemical debris effects on the ECCS for PWRs.

The staff has not issued comparable guidance to BWR licensees (or applicants). For STP Units 3 and 4, the applicant's principal approach to chemical effects is to exclude the materials most likely to be chemical debris sources. Testing in the GSI-191 Program for PWRs identified several insulation materials and other materials as key contributors to chemical effects—including aluminum, calcium silicate insulation, and phosphate pH buffer.

The generation of chemical debris in the water chemistry that is representative of a BWR post-LOCA environment has not been thoroughly studied. Because chemical debris generation may depend on the pH, it is important to consider all sources of acids and bases. Examples include sodium pentaborate potentially added for reactivity control, cesium hydroxide produced by fission, hydrochloric acid generated by the radiolysis of cable insulation, and nitric acid generated by the radiolysis of water and air. The applicant also provided the NRC with access to proprietary chemical effects test results performed by Toshiba Corporation for the RJABWR (ML093090336). The staff determined that this information did not thoroughly address the potential chemical effects in accordance with RG 1.82, because it did not address all potential environmental conditions. Therefore, the staff issued several related requests for additional information to better understand the applicant's evaluation of chemical effects, and how the materials in the operating plant are bounded by the testing and analysis (**RAIs 06.02.02-9, -11, -12**). These RAIs requested the following:

- An explanation of how the chemical effects evaluation was comprehensive with respect to all of the potential combinations of design materials, latent debris, acids and bases, and temperatures.

- Test data and analyses used to support the chemical effects evaluation.

- Demonstration that materials important in debris generation (e.g., aluminum) will not exceed the limits imposed in the licensing basis.

The applicant responded to these RAIs in letters dated September 28, 2009 (ML092730448); December 22, 2009 (ML093580193); January 28, 2010 (ML100330402); and February 22, 2010 (ML100560113).

The applicant's February 22, 2010, response describes calculations performed to support the assumption of no chemical precipitation. This response also includes a proposed revision to FSAR Section 6C.3 stating that because aluminum is prohibited in purchase specifications, no aluminum is expected to be used in the containment. However, the proposed revision includes an assumption that 4.5 square feet of aluminum are present as latent debris (i.e., trash). The applicant determined this surface area based on calculations that evaluated aluminum corrosion and precipitation. These calculations assumed corrosion of the aluminum according to the release rate equations in WCAP-16530-NP-A. The staff's audit of these calculations is documented in the audit summary report dated December 22, 2010 (ML103480636). The calculations also compared the total amount of dissolved aluminum to the solubility limit to determine whether the aluminum would remain dissolved or precipitate as solids at the applicable pH and temperature. This part of the evaluation is based on the solubility data in the report, "Aluminum Solubility in Boron Containing Solutions as a Function of pH and Temperature," (Bahn et al 2008) (Ref. 6.2-12). The applicant's approach generated a value for the surface area of aluminum that would, when corroded (dissolved), remain below the solubility limit and would not precipitate as chemical debris for the 30 days following a LOCA.

The corrosion and solubility calculations were performed for pH values between 5.3 and 8.9, which correspond to the LOCA conditions described in DCD Tier 2, Subsection 3I.3.2.3. The calculations were based on a final suppression pool temperature of 35° C and a large enough mass of aluminum to ensure that it was available to dissolve throughout the 30-day period. The value of the 4.5 square feet proposed in the FSAR corresponds to the pH 5.3 condition that corresponds to the lowest aluminum solubility. The responses to **RAIs 06.02.02-9, -11, -12** also state that there will be no calcium, silicon, or phosphate in the insulation in the containment. The responses explain that a portion of the coated concrete on the floor of the upper drywell is in the zone of influence (ZOI), but any dissolution will be inconsequential due to the absence of phosphate and silicon.

Based on the review of the applicant's aluminum analysis (calculations and Response to **RAI 06.02.02-11** Supplemental 2 dated February 22, 2010 [ML100560113]), the staff determined that the response was not complete. The aluminum corrosion calculations and solubility data used to analyze chemical effects were based on boron-containing solutions. These analytical tools do not apply directly to boron-free BWR coolant, and it was not clear to the staff that this approach would be conservative with respect to an assurance of no aluminum precipitation for 30 days. With respect to concrete, the staff found the RAI responses incomplete because concrete dissolution can generate elements that affect chemical precipitates (e.g., sodium aluminum silicate). In addition, the staff determined that the applicant's analyses did not adequately consider all relevant chemical debris sources. Therefore, the staff issued **RAI 06.02.02-27** asking the applicant to justify the relevance of the aluminum solubility data to the ABWR environment, clarify the post-LOCA pH profile, and address potential chemical effects from other sources such as zinc and concrete. This RAI was tracked as **Open Item 06.02.02-27** in the SER with open items.

The applicant's response to RAI 06.02.02-27 dated June 10, 2010 (ML102090254), does not address all of the staff's concerns about the potential chemical effects, and the staff requested additional information through RAIs 06.02.02-28 to -31. These RAIs focused on additional details about the post-LOCA pH values, the long-term solubility of aluminum, the form of chemical debris that could result from corrosion of inorganic zinc coating, and the possibility of erosion-generated concrete particulate debris.

In the response to RAI 06.02.02-28 dated November 15, 2010 (ML103230133), the applicant explains that the pH of the suppression pool following a LOCA will be in the range of 5.3 to 8.9, but the exact values as a function of time will depend on the actions taken by the STP Technical Services Center staff in accordance with emergency operating procedures. These actions are based, in part, on the initial pH and the post-LOCA sampling results. The licensing basis pH range of 5.3 to 8.9 is specified in DCD Tier 2, Subsection 3I.3.2.3. The applicant also states in the RAI response that sodium pentaborate may be added as a pH buffer from the standby liquid control system to prevent the pH from falling below 5.3. In that case, the applicant's calculations predict that the pH will stabilize in the 8.4 to 8.6 range.

The applicant analyzed chemical effects from concrete, aluminum, and zinc based on the pH range of 5.3 to 8.9. In response to RAI 06.02.02-30 dated November 23, 2010 (ML103510084), the applicant explains that the calculated contribution of concrete and latent aluminum to chemical effects used the WCAP-16530-NP-A methodology. As stated above, the assumed aluminum quantity was 4.5 square feet. The exposed concrete surface was 302 square feet based on the URG value for the amount of qualified coatings assumed to be destroyed on the containment wall by the LOCA jet.

The applicant conservatively assumes that all of the dissolved aluminum and silicon released from the concrete and latent aluminum form precipitates according to the WCAP-16530-NP-A formula. This assumption is conservative because the solubility data support no precipitation, at least for some period of time. The conducted analysis at pH levels of 5.3 and 8.9 provide the minimum and maximum release rates, respectively, over this pH range. It is appropriate to assume precipitation because there is variability in precipitation in the GSI-191 testing program, and because the solubility data for GSI-191 may not apply directly to the STP Units 3 and 4 post-LOCA water chemistry. The staff finds this approach acceptable because the aluminum and concrete release was calculated over a pH and temperature range covered by the WCAP methodology, and assuming the precipitation of all dissolved aluminum conforms to the staff's March 2008 guidance.

One uncertainty in the applicant's aluminum analysis is the calculation of the aluminum dissolution using equations based on data for dissolution in borated water. The staff accepted this as a reasonable approximation because if the solubility of aluminum is higher in borated water (Bahn et al 2008) (Ref. 6.2-12), then the corrosion rate is also likely to be higher. In addition, the STP Units 3 and 4 post-LOCA suppression pool will be borated if the SLC system is added, thus increasing the similarity between the BWR environment and the water used for determining the WCAP corrosion rate equation in the alkaline range. The staff also notes that the applicant's assumption that all dissolved aluminum forms a precipitate addresses some of the uncertainty in the amount dissolved.

The WCAP-16530-NP-A evaluation predicted the formation of aluminum oxyhydroxide (AlOOH) and sodium aluminum silicate ($NaAlSi_3O_8$) precipitates from the elements released from the concrete and aluminum. At a pH of 5.3, the predicted amounts of AlOOH and $NaAlSi_3O_8$ were 48 grams (g) and 6 g, respectively. At a pH of 8.9, the predicted amounts were 298 g and 6 g, respectively.

In the response to RAI 06.02.02-29 dated November 23, 2010 (ML103510084), the applicant provides an analysis of the assumed 47 pounds of IOZ coating as a source of chemical debris. The analysis postulated that the entire 47 pounds (as 604 square feet) of coating would be reduced to its constituent particles by the LOCA jet. In addition, the particles were assumed to be 100 percent metallic zinc, which is conservative because IOZ coatings are typically 80 to 90 percent zinc by weight according to literature and product data sheets. The particles were assumed to be 10-micrometer spheres, which the staff accepted for the coating particle size in the SER for Nuclear Energy Institute (NEI) 04–07 (ML043280641). The applicant calculated a total surface area of more than 22,600 square feet, which was used as the input in the WCAP-16530-NP-A evaluation at pH levels of 5.3 and 8.9. At a pH of 5.3, all 47 pounds (21.3 kilograms [kg]) of the zinc were dissolved. At a pH of 8.9, the WCAP-16530-NP-A calculation predicted 4.96 kg of dissolved zinc.

The applicant's analysis assumes that the WCAP release rate equation for zinc applies to the STP Units 3 and 4 post-LOCA suppression pool environment. The staff found this assumption reasonable given that general descriptions of zinc corrosion in the literature identify a strong pH dependence, but they do not identify boron or borate as a corrosion inhibitor. In addition, at a pH of 5.3, the applicant's analysis is bounding (100 percent corrosion of the zinc); and at a pH of 8.9, the water will be borated and more like the water used for determining the WCAP corrosion rate equation. The staff also noted that the assumed total destruction of all of the coating increased the surface area, and thereby the zinc release, by a factor of approximately 40.

Although the WCAP-16530-NP-A spreadsheet calculates a zinc release, it does not convert the dissolved zinc to a solid (precipitate), because the incremental contribution of zinc is considered negligible for PWRs. Therefore, for STP Units 3 and 4, the applicant assumes that all of the dissolved zinc will precipitate as a gelatinous zinc oxide (ZnO). The staff found the assumption of ZnO reasonable because the literature predicts ZnO and $Zn(OH)_2$ as the stable corrosion products of zinc in water over the pH and temperature ranges of interest, and there are no data indicating the formation of other zinc-base precipitates in the ABWR post-LOCA water chemistry. The staff considered it conservative to assume that all of the precipitate is gelatinous, since the literature reports that the zinc corrosion product in pure water may be gelatinous, granular, flaky, or dense between 20°C and 100°C (Ref. 6.2-13)..

In order to determine the licensing basis chemical debris quantity, the applicant compared the total debris at each pH value and selected the higher debris total as the licensing basis. The higher debris total was at pH 5.3, with the following calculated debris quantities:

$AlOOH$	0.048 kg (0.10 lb)
$NaAlSi_3O_8$	0.006 kg (0.01 lb)
ZnO	26.6 kg (58.6 lb)

The staff reviewed the basis for this release of chemicals and precipitation of debris, including the applicant's assumptions and inputs to the WCAP-16503-NP-A spreadsheet. This review was conducted through an audit (ML103480636) of the applicant's calculation note. The staff also used the applicant's assumptions, alternative assumptions, and the temperature profile in the technical report to perform independent confirmatory calculations using the WCAP-16530-NP spreadsheet. The staff also performed the calculations manually using the release rate equations derived in WCAP-16530-NP-A and incorporated into the spreadsheet. The staff's calculations were in agreement with those of the applicant's.

In the response to RAI 06.02.02-31 dated October 14, 2010 (ML102910232), the applicant explains that the concrete analysis does not include particle generation by the LOCA jet. This is based on the proximity of the LOCA jet and the calculated pressure at the concrete surface, along with proprietary test data on the effects of subcooled jets on concrete (WCAP-7391 "Pressurized Water and Steam Jet Effects on Concrete", 1970). The staff found this justification acceptable for eliminating concrete erosion as a chemical effects debris concern, because the pressure of the LOCA jet closest to the concrete surface for STP Units 3 and 4 is much lower than the minimum pressure in the WCAP-7391 testing that caused no damage. The staff also noted that according to the WCAP-16530-NP-A formula, large increases in the concrete surface area (e.g., from the generation of concrete particles) would add a negligible amount to the applicant's proposed chemical debris load.

The applicant assumes that all of the chemical debris will transport to the suction strainers but will cause minimal head loss. The applicant's responses to RAIs 06.02.02-29 and -30 explain that there will be minimal head loss, because the small amount of latent fiber will not generate a continuous bed on the strainers and the calculated chemical debris quantity is small relative to the strainer area. The staff found this assumption acceptable because it conforms to the staff's March 2008 chemical effects guidance. This guidance states that plants may use a simplified chemical effects evaluation in the case of a bare strainer area, because chemical precipitates are expected to pass through the strainer.

The applicant assumes that all of the fibers and chemical debris pass through the suction strainers and include the 26.6 kg (58.6 pounds) of chemical debris in the consideration of downstream effects. As stated in the responses to RAIs 06.02.02-29 and -30, the applicant will include the chemical debris in the test plan for fuel assembly testing, which is described in FSAR Appendix 6C and in the response to RAI 04.04-4 (ML110100696). The staff found this acceptable because the chemical debris is assumed to proceed downstream after passing through the suction strainers. FSAR Subsection 6C.3.1.9.3 lists the debris sources for the fuel assembly test and describes how the quantity will be scaled down for the test. In the RAI responses and in FSAR Subsection 6C.3.1.9, the applicant states that AlOOH will be used as a surrogate for all chemical debris, including the zinc oxide; and it will be prepared according to WCAP-16530-NP-A.

The staff found the use of the AlOOH and $NaAlSi_3O_8$ as precipitates acceptable, because the staff's SE on WCAP-16530-NP-A states that a surrogate precipitate prepared in accordance with the directions in WCAP-16530-NP-A provides adequate settlement and filterability characteristics to represent post-LOCA chemical precipitates in strainer head loss tests. The staff's limitations and conditions in the safety evaluation of WCAP-16530-NP-A include surrogate debris settlement testing and acceptance criteria. PWR licensees and applicants have used this approach in strainer and fuel assembly tests, but because zinc precipitates were not addressed in GSI-191 head-loss testing, the staff had not previously considered a zinc surrogate. However, testing at Argonne National Laboratory found that with a fiber bed, the pressure drop capacity of a strainer test loop was exhausted almost immediately using a WCAP-16530-NP-A surrogate quantity equivalent to 5 parts per million (ppm) dissolved aluminum. Even a surrogate addition equivalent to 1.5 ppm dissolved aluminum exceeded the pressure drop capacity. Given the effectiveness of the AlOOH surrogate at causing head loss in combination with a fiber bed, the staff concluded it was possible, but unlikely, that dissolved zinc would form a more potent chemical. Therefore, the staff found the applicant's use of the AlOOH surrogate acceptable.

For the reasons discussed above, the staff found that the applicant's assessment of chemical effects conforms to RG 1.82, Revision 3 and also conforms, or provides reasonable alternatives, to the staff's March 2008 guidance on GL 2004-02 with respect to the type, quantity, and head-loss testing of postulated chemical debris. Therefore, the staff concluded that the applicant meets the requirements of 10 CFR 50.46, as they relate to chemical debris effects on the ECCs.

The evaluation of ECCS components downstream of the suction strainers is meant to address blockage of flow paths, wear, and abrasion (e.g., valves, pumps, components, and heat exchanger tubes), and blockage of fuel assembly flow channels. In RG 1.82 Revision 3, Subsection 2.1.2.2 addresses the need to prevent the clogging of flow restrictions and damage from fine particles downstream of passive strainers. In an SE dated December 2007 (ML073520295), the staff accepted, with certain limitations and conditions, the methodology and acceptance criteria described in WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191." In the response to **RAI 06.02.02-1** dated April 29, 2009 (ML091270491), the applicant proposes to evaluate downstream components for STP Units 3 and 4 in accordance with WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191." The applicant states that this evaluation will be performed during the detail design phase of the plant. The staff reviewed the applicant's proposal to assess the applicability of WCAP-16406-P to STP Units 3 and 4 and the conformance to the staff's accompanying SER. Conforming to this guidance provides one acceptable way to meet the requirements of 10 CFR 50.46, as it relates to downstream effects on components.

Because WCAP-16406-P was developed for PWRs, the staff issued **RAI 06.02.02-10** asking the applicant to justify applying that methodology to a BWR. In the response to **RAI 06.02.02-10** dated September 28, 2009 (ML092730448), the applicant states that the WCAP-16406-P methodology is applicable to STP Units 3 and 4 based on the similar materials and clearances for BWR and PWR downstream components. In the response to **RAI 06.02.02-13** dated December 21, 2009, (ML093580193), the applicant states that the analysis will be performed using the acceptance criteria in the WCAP and the accompanying NRC staff safety evaluation (Ref. 6.2-13). The response included a corresponding revision to FSAR Section 6C.3.2. The staff's review focused on the applicability of WCAP-16406-P to the analysis of downstream effects.

The staff compared WCAP-16406-P to ABWR design information to confirm that the WCAP addresses the types of components and materials used in the ABWR ECCS. The ABWR DCD states that ECCS pumps are centrifugal pumps and valves are conventional gate, globe, and check valves for nuclear service. Section 5.4 of the ABWR DCD states that RHR pumps are centrifugal pumps and the valves are conventional gate, globe, and check valves for nuclear service. Also, RHR heat exchangers are the shell and tube type. DCD Table 6.1-1 indicates that the valve and heat exchanger materials are conventional carbon and stainless steels, which are the same as or similar to the materials listed in the WCAP report. Because these component types are addressed by WCAP-16406-P, the staff determined that it is appropriate for the applicant to use that methodology to analyze downstream components. The basis for the staff's acceptance of WCAP-16406-P is discussed in more detail below for each of these component types.

Centrifugal pumps for ECCS are covered in WCAP-16406-P and in the staff's corresponding SE. The analysis considers how the wear of internal pump components affects hydraulic performance (head and flow), mechanical performance (vibration), and pressure boundary performance (shaft seal integrity). In the SER on the WCAP report, the staff found the pump evaluation methodology acceptable based on its use of conservative assumptions, a standard engineering evaluation, and consistency with the staff's SER on NEI 04-07. The staff's SE on WCAP-16406-P also identified limitations and conditions related to ECCS pumps, such as confirmation that the assumed mission time of 720 hours in the WCAP bounds the plant's mission time.

As stated above, valves in the ABWR ECCS are conventional gate, globe, and check valves for nuclear service. WCAP-16406-P includes a wear evaluation for all valves and a description of the significance of wear on each valve type and size. The evaluation includes a calculation of the flow area increase due to wear. For valves considered to be critically sensitive to wear, the WCAP requires the licensees to contact the manufacturer for a flow effect assessment, if the evaluation indicates that wear causes a flow area increase of more than 3 percent. NRC determined in the SER on the WCAP that the steps in the evaluation are acceptable because they are based on standard engineering practices. The SER also found the value of the 3 percent flow area acceptable, because it is within valve manufacturing and fluid-flow calculation tolerances.

All valves with pipe diameters greater than 1.5 inches, (and nearly all valves smaller than that), are evaluated for plugging. Some valves are in the closed position during the postulated accident and therefore require no plugging evaluation. The evaluation guidance is according to valve type and size, with vendor input required to determine the flow area for certain valves. The staff found this guidance acceptable in the SE for WCAP-16406-P because it conforms to the staff's SER for NEI 04-07. To summarize for valves, because WCAP-16406-P requires plugging and wear evaluations for all valves, except those in the closed position during the event, the WCAP-16406-P analysis is applicable to STP Units 3 and 4.

For shell and tube heat exchangers, the WCAP addresses both wear and blockage. The WCAP addresses wear through standard industry methods for evaluating the consequences of tube wall thinning. For tube plugging, the WCAP states that a plant-specific evaluation is needed if the inside diameter of the tube is less than the size of the largest expected debris particle. The staff found this approach acceptable in the SER for the WCAP, because the wear evaluation conforms to standard industry practice and particles smaller than the tubes should not cause blockage. The WCAP states that debris settling in heat exchangers is not a concern, based on the expected velocity exceeding the settling velocity. However, the staff's SER stated that licensees should confirm flow velocity and should evaluate heat exchanger plugging if the velocity is less than the settling velocity.

For the reasons discussed above, the staff determined that WCAP-16406-P is applicable to the evaluation of downstream components for STP Units 3 and 4. Therefore, the applicant conforms to RG 1.82 Revision 3, Subsection 2.1.2.2 related to downstream components by meeting the acceptance criteria in the WCAP. In addition to proposing a revision to FSAR Section 6C.3.2, the applicant's response to **RAI 06.02.02-11** also adds a new commitment, COM 6C-1, to submit the analysis to the NRC at least 18 months before fuel loading. The proposed commitment states the following:

> Downstream effects analysis for components (pumps, valves, and heat exchangers) will be performed in accordance with WCAP-16406-P and the accompanying SER, and the evaluation submitted to the NRC.

The commitment to provide the evaluation 18 months before fuel loading is acceptable because the evaluation will be available for the staff's verification before fuel loading. The proposed revisions to FSAR Section 6C.3.2 are acceptable because the description of the evaluation of downstream components is consistent with WCAP-16406-P and the staff's accompanying SER, and because the description identifies the commitment and timing for completing the evaluation.

As stated in the commitment, the applicant will perform the WCAP-16406-P analysis in accordance with the staff's accompanying SE. Therefore, in accordance with COM 6C-1, the applicant will need to review the staff's evaluation of the WCAP and determine whether limitations and conditions apply. The proposed revisions, including FSAR Section 6C.3.2 and COM 6C-1, were tracked as **Confirmatory Item 6.02.02-27**. The staff verified that Revision 4 of FSAR Section 6C.3.2 and COM 6C-1 reflect the changes discussed in the response to RAI 06.02.02-27. This revision resolves Confirmatory Item 06.02.02-27.

In providing bypass debris quantities for downstream fuel effects testing in the response to RAI 04.04-3 Supplement dated February 22, 2010 (ML100560113), the applicant states the assumption that 100 percent of the sludge, dust/dirt, and rust flakes will pass through the ECCS strainers and reach the RPV. This assumption is conservative because the applicant does not take credit for sludge, dust/dirt, or rust flakes settling in the suppression pool or on the ECCS strainers.

However, in the February 22, 2010 response, the applicant states the assumption that only 10 percent of 1 cubic foot of latent fiber and 2 percent of RMI within the break zone of influence will pass through the ECCS strainers. The applicant bases the 10 percent of latent fiber bypass fraction on CCI-cassette-type strainers testing. Considering the small amount of latent fiber present compared to the size of the STP Units 3 and 4 ECCS strainers, during a public teleconference on July 14, 2010 (ML102090236), the staff requested the applicant to justify the applicability of CCI bypass testing. The applicant's response to RAI 04.04-4 response, dated January 6, 2011 (ML110100696), states the assumption that 100 percent of the latent fiber will pass through the ECCS strainers. This assumption is conservative because the applicant does not take credit for the settling of any latent fiber in the suppression pool or on the ECCS strainers.

The applicant's February 22, 2010, response to RAI 04.04-3 identifies Figure 3-7 of NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," as the basis for assuming that only 2 percent of the RMI within the break zone will pass through the ECCS strainers. According to the staff, this figure shows that the smallest sized group that generated the RMI shards was less than 6.4 millimeters (mm) (1/4 inches [in]), which comprised 4.3 percent of the total amount generated by a lager pipe break. During a public teleconference on July 14, 2010 (ML102090236), the staff requested the applicant to provide the basis for assuming that only 2 percent of the RMI within the zone of influence was assumed to be less than 2.1 mm and would pass through the ECCS strainer. The applicant's response to RAI 04.04-4, dated January 6, 2011, states that 4.3 percent of the RMI within the break zone of influence will pass through the ECCS strainers. This assumption is conservative because only a fraction of this 4.3 percent of the RMI in the smaller than 6.4-mm group will be smaller than 2.1 mm and will pass through the ECCS strainers.

The applicant agrees to update STP Units 3 and 4 FSAR Subsection 6C.3.1.9 accordingly to provide quantities and justification for debris assumed in downstream fuel effects testing. This agreement is being tracked as **Confirmatory Item 04.04-4**.

Therefore, as discussed above, the staff determined that the applicant has conservatively estimated the amount of debris (coatings, sludge, dust/dirt, rust flakes, latent fiber, RMI shards, and chemical debris) that would pass through the ECCS strainers for use in the downstream fuel effects testing.

The ECCS strainer design also affects core cooling. During the post-LOCA containment recirculation long-term core cooling phase, the containment debris and chemicals downstream of the strainer can enter the reactor vessel and potentially cause a core flow blockage and plate-out of chemical precipitates on the fuel cladding, thus resulting in a degradation of the core heat transfer. FSAR Appendix 6C presents an in-vessel evaluation that assesses the impact of debris in the post-LOCA recirculating water on components inside of the reactor vessel, including the core inlet and fuel assemblies.

In RAI 06.02.02-2, the staff requested the applicant to describe how the downstream effects identified in RG 1.82, Revision 3 will be addressed. The applicant's response to RAI 06.02.02-2 dated September 28, 2009 (ML092730448), states that the ABWR design as applied to the STP Units 3 and 4 plants provides reasonable assurance that downstream effects, as a result of debris bypassing the ECCS suction strainers, will not have a deleterious effect on critical components such as fuel rods, valves, and pumps downstream of the suction strainers. This reasonable assurance is based on the following (as described in the September 28 response to RAI 06.02.02-2):

- The relative reduced likelihood of latent debris generation compared to operating BWRs and PWRs (restricted access to the containment, the suppression pool cleanup system, the operational program for suppression pool clean-up)

- Minimal LOCA-generated debris (elimination of recirculation piping, no fibrous insulation)

- Reduced impacts of chemical effects

- ABWR design features that minimize the transport of accident-generated debris

- Suction strainer design

- Diversity of ECCS delivery locations

The applicant also agrees to a COL license condition to perform a test confirming that the fuel for the initial fuel load meets the specified acceptable fuel design limits, as required by GDC 10, considering the downstream effects of containment debris on the reactor fuel. The staff agreed that the design features of the ABWR will provide reasonable assurance, but the staff also agreed that a fuel test is required to verify that the debris loading will not negatively affect core cooling. The staff determined that the acceptance criterion specified in the response was not sufficient, because it did not include specific verifiable parameters for test acceptance. The staff issued RAI 04.04-3 requesting the COL applicant to (1) provide a verifiable criterion for the fuel testing, (2) revise FSAR Section 4.4 to include the details of the acceptance criterion, and (3) confirm that the protective coatings debris characteristics for fuel assembly tests will be generally consistent with NRC guidance for operating PWRs. The applicant submitted the response to RAI 04.04-03 and the RAI supplement in letters dated January 18, 2010 (ML100191526), and February 22, 2010 (ML100560113). The applicant states that: "STP 3 & 4 design unique testing will be performed to confirm that downstream effects will not impair the provision of adequate flow to provide long term cooling for the fuel. Acceptance criterion for this testing will ensure adequate flow rate through the core region to cool the fuel for an extended period of time post-LOCA." The applicant prepared the calculation determining the acceptance criterion for the proposed license condition and made the calculation available to the staff for audit. The resolution of this issue was tracked as **Open Item 04.04-3** in the SER with open items.

Audit Activities

As a followup to the review, the staff audited the Westinghouse calculation for determining the acceptance criterion on July 12; September 28; and December 7, 15, and 20 of 2010 (ML110030827). Initially, the applicant stated that the proposed acceptance criterion for the license condition will be a pressure drop across the fuel assembly inlet less than the value of 5 psid, which determined through an analysis to provide adequate long-term cooling. The purpose of the staff's audit was to review the supporting STP documentation for the downstream fuel effects test calculations, in order to confirm that the downstream fuel effects license condition acceptance criterion is appropriate. During the audit, the staff noted that the calculations supporting the downstream fuel effects test acceptance criterion of 5 psid were based on the SVEA-96 Optima II (Optima-2) fuel rather than on the GE 8 x 8 fuel approved in the ABWR certified DCD. As a result of the audit, the staff issued RAI 04.04-4 requesting the applicant to explain how the proposed acceptance criterion is suitable for the fuel design that is currently the basis for the ABWR design. The applicant's response to RAI 04.04-4 dated January 6, 2011 (ML110100696), revises the downstream fuel effects license condition acceptance criterion and the calculation supporting that acceptance criterion. The acceptance criterion is based on an equation that includes a K-loss factor of 1,200 to replace the differential pressure value of 5 psid. The applicant made the new calculation available for the staff to audit on September 28, 2010. The applicant gave a presentation summarizing the proposed revision of the license condition acceptance criterion and the calculation supporting that license condition. The Westinghouse presentation, "Revision to STP 3/4 Analysis of Downstream Debris Effects on Fuel," can be found in ADAMS at Accession No. ML102870255. The staff reviewed the calculation during the audit and asked several clarifying questions. The applicant formally responded to several of the staff's clarifying questions in a letter dated October 25, 2010 (ML103010274).

On January 6, 2011, the applicant submitted the response to RAI 04.04-4. The applicant's response provides the FSAR revision that describes the approach, results, and test plan of the analysis. The response also explains how the applicant's calculation bounds the fuel certified in the DCD. Furthermore, the response provides the revised proposed license condition for the fuel testing. As stated above, the revised license condition provides an equation correlating differential pressure and flow rate for clean and fouled conditions.

Test Acceptance Criterion Analysis

In FSAR Appendix 6C.3.1.9.1, the applicant states that the analysis determines the acceptable level of blockage in the fuel by LOCA-generated debris, which bypasses the ECCS suction strainer. This analysis ensures compliance with 10 CFR 50.46(b)(5), in which the long-term core cooling is maintained; the calculated peak cladding temperature is maintained at an acceptably low value; and the decay heat is removed for an extended period of time that is required by the long-lived radioactivity remaining in the core. The potential deposition of particulate, chemical effects, and fibrous debris on the fuel and its impact on the heat transfer from the cladding is also included in the evaluation. The results of the analysis are used to determine the acceptance criterion for the downstream fuel effects test to be performed at least 18 months before initial fuel loading. The acceptance criterion for this test is based on the following equation:

$$(P_f / \quad {}_i)^2 \leq 1200 \text{ x } (w_f / w_i)^2$$

Where: subscript "i" denotes initial (i.e., unfouled) conditions, and "f" indicates fouled conditions, w is the flow rate into the assembly, and Δp is the pressure drop from the bundle inlet to downstream of the third grid.

For the test acceptance criterion calculation, the applicant used staff-approved Westinghouse BWR LOCA computer codes (GOBLIN and CHACHA-3D), which are described in Topical Report WCAP-16078-NP-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96-Optima 2 Fuel," November 2004 (ML050390435). The analysis models the actuation of automatic features such as the main steam isolation valve closure, reactor scram, and the ECCS. This analysis also determines the boundary conditions that are applied to the hot channel analysis.

Although the staff has approved these codes for use in operating BWR/2 through BWR/6 plants using the Westinghouse BWR fuel, the staff has not approved these codes for application to the ABWR design. However, the staff believes they can be utilized for this analysis because the thermal and hydraulic phenomena to be modeled are the same as in conventional BWRs, and both codes have been shown to adequately predict these phenomena. The unique features of the ABWR can be addressed by code inputs based on an ABWR model that reflects these design differences.

The ABWR LOCA is assumed to occur coincident with a loss of offsite power, and it is different from a typical BWR LOCA in two important ways:

- The core flow rate decreases quickly due to the rapid coastdown of the RIPs following the loss of power.

- The elevations of potential large pipe break locations are above the top of the active core.

The first difference results in an early boiling transition before the reactor scram occurs. The reduction in the heat transfer results in an increase in the fuel cladding temperature. The decrease in core power caused by increased voiding and reactor scram results in a rapid reduction in the cladding temperature. As a result, the cladding temperature excursion is short-lived. The second difference in conjunction with the actuation of the ECCS, results in a nearly continuous two-phase cooling of the core. The typical extended core uncovery phase of the BWR LOCA does not occur in the ABWR. As a result, the peak cladding temperature occurs before the ECCS actuation and is independent of the ECCS performance.

The calculation was prepared in order to establish the predicted behavior for the pressure drop and flow resulting from the imposed fuel blockage. The purpose of the calculation was to determine the maximum level of blockage that would allow the core to maintain a void faction of 0.95 or less. The applicant also performed a calculation to determine the effect that fouling thickness would have on the peak cladding temperature. In reviewing the calculation package, the staff examined the applicant's key assumptions to determine the conservatism. Specific items reviewed by the staff are discussed in the following paragraphs.

Break Selection

In the ABWR design, all penetrations to the RPV are above the active core. Because the ABWR uses RIPs, the large recirculation piping characteristic of conventional BWRs is eliminated. The ABWR breaks are postulated to occur in much smaller diameter lines than conventional BWR recirculation inlet or outlet lines. The applicant determined that the FWLB inside the drywell is the limiting case for fuel thermal and mechanical performance because the downcomer level, which can be maintained by the ECCS, will be lower than for a MSLB. This is due to the placement of the feedwater spargers at a lower elevation than the Main Steam nozzle. The static head of the downcomer, which is essential for maintaining a natural circulation path through the core, will be lower for the FWLB. The staff agreed that this assumption is conservative, since the maximum downcomer water level that can be achieved corresponds to the break elevation. Any excess ECCS liquid added will spill directly out the break and will not contribute to the static head.

Analysis Model

The GOBLIN model for the ABWR reactor vessel and connected systems is similar to conventional BWR models. The nodalization used by the applicant accounts for the differences in design between BWRs and ABWRs. Most significant is the replacement of BWR external recirculation loops with ABWR internal recirculation pumps. In the ABWR design, the peak cladding temperature will occur in the first few seconds of a LOCA transient, since a rapid coastdown of all RIPs is assumed. This was demonstrated during the original design certification evaluation of the ECCS performance. The case with the highest peak cladding temperature predicted by the analysis was further analyzed in a lattice heatup calculation using CHACHA-3D, the fuel rod mechanical and thermal response computer code.

The GOBLIN model considers the hot channel with a power peaking factor of 1.7 and the average channel, which represents the remainder of the core. All bypass flow paths are considered closed. Other model assumptions include:

- Using a 95 percent void as the acceptance criterion for demonstrating the satisfaction of the thermal and mechanical operating limits of the specified acceptable fuel design limits (SAFDLs) (GDC 10 requirement).

- Applying a significant flow area reduction for more than 1 minute beginning at 850 seconds after break initiation, due to debris blockage.

- No credit for the HPCF.

- No credit for the bypass flow paths.

- Two trains of the low-pressure core flooder (LPCF).

- Decay heat held constant at 3.2 percent of the rated thermal power beginning at 5 minutes into the LOCA.

- Debris introduced when the quasi-steady-state is reached (850 seconds after break initiation).

Detailed Model Description

- Fuel details are based on the Westinghouse OPTIMA-2 design.
- Integral debris filter.
- Uniform crud thickness of 30 microns.

The 95 percent void conditions are predicted by the GOBLIN code at the beginning of the transition from nucleate to film boiling in the upper part of the hot bundle. By definition, the minimum critical power ratio for a fuel rod corresponds to the boiling transition and is specified to ensure that the SAFDLs are satisfied. Because the void fraction will be maintained below 95 percent, there will be no boiling transition. Therefore, the staff believes that this will ensure that all fuel thermal and mechanical design limits will be met and is therefore an adequate indication of acceptable fuel performance.

The staff compared the GOBLIN model nodalization to the model utilized in WCAP-16078-P-A. The approved model applies the 1971 ANS decay heat curve plus 20 percent. The modified GOBLIN model conservatively applies a constant heat load at 5 minutes. The integrated heat addition to the fuel rods will be significantly larger than would actually occur, if allowed to decay exponentially.

The approved analyses credit flow into the top of the fuel bundles from one train of core spray. The modified GOBLIN analysis results show that the acceptance criterion (less than 95 percent void conditions) can be met without crediting the HPCF, which functions in a manner similar to conventional BWR core spray. The HPCF is an additional defense-in-depth system, which will further ensure that all fuel design criteria will be met for the long-term cooldown.

The approved analyses credit the normal bypass flow paths, while the GOBLIN model used for the calculation does not. The staff agreed with the applicant that this conservatively minimizes alternate pathways for the coolant to enter the fuel channels to replace the coolant mass, which boils off and exits the reactor vessel through the break.

To investigate the effects of a reduction in flow area due to the buildup of debris, the applicant made a series of GOBLIN runs varying only the hot channel and average channel inlet areas. The GOBLIN model treats the flow path as a sudden contraction. The flow loss associated with the area is equivalent to that of a sharp-edged orifice in an incompressible fluid piping system. Although the effective loss of the debris bed is unknown, the staff believes that the modeling approach is conservative and acceptable because of the multiple conservatisms taken by the applicant. The staff also found that the K-loss factor of 1,200 is a reasonable approximation. The actual K-loss factor will be confirmed during the test. The staff concurred with the approach presented by the applicant to show that the K-loss factor used will allow for adequate core cooling. The staff concluded that the calculation, coupled with the multiple conservatisms the applicant is taking with regard to the fuel test, demonstrate that the applicant's acceptance criterion proposed in the response to RAI 04.04-4 dated January 6, 2011, is conservative. Therefore, the staff concurs that adequate core cooling will be maintained if the test demonstrates measured losses of less than 1,200.

Analysis Approach

The STP Units 3 and 4 ECCS consists of two HPCF systems that inject inside the core shroud through spargers at the top of the core; three LPCF systems (one injecting through the feedwater sparger and the remaining two spargers injecting through the annulus region of the core, thus taking suction from the suppression pool); and one RCIC system.

Although the diversification of ECCS delivery points (injection from the top of the core by the HPCF and injection from below the core by the LPCF and RCIC) helps to reduce the consequences of a blockage in the fuel assembly, the FSAR describes the applicant's analysis as assuming that all of the debris is injected from the bottom of the core and therefore passes through the fuel debris filter inlet, which is the most likely place for the blockage to occur. Following the break and after the blowdown is complete, the water level in the downcomer rises to the feedwater sparger (i.e., the break elevation). At that point, all of the excess flow from the LPCF or the RCIC not injected into the core will flow out through the break. The applicant then states that the flow rate into the core is dependent on the natural circulation head of colder water in the downcomer and the hotter water and two-phase mixture in the core region. As the core inlet begins to block, the core flow rate decreases.

For this analysis, the applicant reduced the flow area at the fuel inlet to simulate blockage of the fuel assembly inlet. The applicant also blocked all bypass flow paths except for the inter-assembly bypass holes located in the bottom transition piece. The applicant pointed out that the bypass in the bottom nozzle is not likely to be blocked due to the large opening size (10.3 mm in diameter), which is significantly greater than the strainer hole size (2.1 mm). The staff agreed with this assumption. The reduced flow area at the core inlet decreases the core inlet flow rate and increases the core inlet differential pressure. The applicant determined the minimum flow area to ensure that no point in the core experiences significant cladding heatup, which is measured by ensuring that the void fraction remains less than 0.95. The applicant used the corresponding differential pressure at the core inlet (which is the parameter to be monitored in the test) and corrected for the changes in the core flow rate as the acceptance criterion for the test.

The applicant chose values of the nodal power peaking and pin-to-pin peaking factors for the hot assembly to place the hot rod at the thermal mechanical operating limit. The applicant assumed a core power corresponding to the decay heat at 5 minutes after shutdown, as the debris accumulates at the fuel assembly inlet and reduces the inlet flow area.

The applicant conservatively kept the core power corresponding to the decay heat at 5 minutes constant thereafter. In FSAR Appendix 6C, the applicant states that a blockage sufficient to reduce core cooling within 5 minutes is not likely for the following reasons:

- The core and the upper plenum retain significant inventory during the blowdown. The void fraction in the upper plenum remains below 1.0. Therefore, additional water injected into the core before a quasi-steady-state is established is minimal (i.e., the level in the downcomer increases to the feedwater line). After the quasi-steady-state is achieved, the injection into the core is limited by the natural circulation head and core boil-off.

- The debris-laden flow from the suppression pool will be injected into the vessel only after the initial inventory of the ECCS piping, which is clean, is swept and injected into the vessel. Therefore, any suppression pool water will be further diluted by this clean initial injection.

- Although not credited in the applicant's analysis, the HPCF (and RCIC) pumps initially inject from the condensate storage tank, which is a clean source of water. The LPCF pumps do not start injection until well after 2 minutes. In addition, a parametric study is performed to determine the effect of fouling caused by the deposition of particulate, fibrous, and chemical effects debris on the cladding. The level of initial fouling on the cladding is increased to represent the effect of the uniform deposition of particulate debris on the cladding.

Analysis Results

The applicant's supplemental response to RAI 04.04-4 dated January 18, 2011 (ML110190798) provides five figures that compare the core inlet differential pressure, flow rate, void fractions, and peak cladding temperature for the cases with no blockage and for cases with blockage that result in a reduction of flow area by a significant percentage of the inlet flow area. The applicant estimates 5 minutes for the time required for debris in the containment to start to reach the fuel filters. The models assume that clogging begins at 850 seconds, because that is when the flow through the core reaches steady-state. In order to ensure that any effects seen are covered by the changes being made, steady-state flow is required. Despite a very high level of blockage, the applicant's analysis shows that sufficient flow remains available to the core to ensure that the core void fraction both in the hot assembly and in the average assembly remain less than 0.95.

In the ABWR design, the peak cladding temperature occurs very early in the transient, during the coastdown phase of the RIPs and before ECCS injection occurs. The peak cladding temperature remains unaffected by the subsequent blockage at the fuel inlet, because the cladding temperature is maintained near the saturation temperature as the core void fraction is maintained below 0.95 in both the hot and average assemblies. Figure 04.04-4-5 in the applicant's supplemental response to RAI 04.04-4 compares the cladding temperature for the blocked and unblocked cases. The low fuel cladding temperature also ensures that cladding oxidation does not occur in the long-term cooling phase of the accident.

The calculations performed by the applicant were based on an SVEA-96-Optima 2 fueled core. In the response to RAI 04.04-4 dated January 6, 2011, the applicant states that the reason the calculations also apply to GE 8 x 8 fuel is because the differences between the GE 8 x 8 and the Optima 2 fuel designs are not significant for purposes of the calculations. Because each assembly in the ABWR is a closed channel, the flow into the assembly required to remove the specified assembly power without transitioning to steam cooling is independent of the specific fuel mechanical design. The flow rate into the assembly is determined by the hydraulic head from the water in the downcomer and the hydraulic characteristics of the fuel assembly. Therefore, hydraulic characteristics such as resistance and bundle flows that are required for removing the specified power from the hot assembly and for preventing steam-only cooling must be similar for any fuel design. Fuel assemblies used in the ABWR must be hydraulically similar in order to meet various design features of the ABWR. CENPD-300-NP-A (Nonproprietary), "Reference Safety Report for Boiling Water Reactor Reload Fuel," dated July 1996 (ML110260388), shows that the hydraulic characteristics of a GE 8 x 8 fuel and an Optima 2 fuel are very similar. The staff reviewed the mechanical design features of both fuel types presented in the referenced topical report and found the explanation to be reasonable. Additionally, a similar argument was applied to operating reactors with mixed cores. Plant operating experience has shown that replacement fuels with similar designs have similar performance characteristics. The same can be expected for the ABWR design. The staff concurred that the applicant's calculation bounds the fuel certified in the DCD.

In response to staff's questions, the applicant response dated October 25, 2010 (ML103010274), states that in the calculation, a blocked inlet loss factor reduction of 4 was selected as a conservative margin to bound fuel design differences and uncertainties associated with extrapolating test conditions to design conditions. Because the loss coefficient varies with the area squared, the factor of 4 reduction in the loss coefficient is equivalent to a factor of 2 for the increase in the flow. The staff concluded that the use of the reduction factor of 4 in the calculation is conservative and will compensate (to some extent) and account for the design differences between the GE 8 x 8 fuel and the Optima-2 fuel designs and uncertainties in the debris bed flow loss.

Fouling

The applicant states that the impact on the clad temperature due to fouling caused by an assumed deposition of particulate debris on the cladding is small. To address the concern that post-LOCA containment debris and chemical precipitates can plate-out on fuel rod cladding and impede heat removal from the fuel rods, the applicant evaluated the impact of post-LOCA debris deposition on fuel rods for the core. The source of debris products is the interaction of the fluid inventory in the post-LOCA suppression pool environment with debris and other materials exposed to and submerged in the pool. The purpose of the evaluation was to predict a maximum scale thickness of the resulting cladding deposit buildup and a maximum clad/oxide interface temperature resulting from the deposits during the post-LOCA recirculation long-term core cooling phase.

The total amount of debris (which consists of paint/coatings, sludge/dirt/dust/rust) that the applicant assumed was taken from the referenced ABWR. The applicant estimated that the thickness of the “crud“ layer, normal values of the fouling, range between 0.0 and 10.0 microns (0.39 mils). The applicant assumed that the debris generated is deposited evenly over the fuel and will generate a layer 14 microns (0.55 mils) thick. The applicant performed a sensitivity study in which 30 microns (1.18 mils) were applied along the length of the fuel. This increased fouling layer was shown to cause an increase of 30° F in the peak cladding temperature. For the calculation, the applicant used the staff-approved BWR LOCA methodology, GOBLIN and CHACHA-3D computer codes. Debris amounts specified for the fuel test include a penalty factor to account for the possibility of nonuniform debris deposition. Using the CHACHA-3D analysis model for calculating the peak cladding temperature the staff concluded that that the estimation of debris deposited on fuel surfaces is reasonable. The staff also agreed that the proposed crud thickness of 30 microns (1.18 mils) provides reasonable assurance that long-term cooling will be successfully maintained.

Test Plan

In FSAR Appendix 6C.3.1.8. 9.2, the applicant proposes a test plan for the downstream fuel effects test. The applicant’s test facility is comprised of a fuel assembly mock-up, a pump, associated recirculation piping, and a mixing tank to add the debris. The test will be conducted with a single partial height fuel assembly that includes an integral debris filter, a fuel inlet nozzle, any integral debris filters, a lower tie plate, and fuel spacer grids. The applicant will model the cross section of the fuel exactly but will reduce the length of the fuel assembly. The fuel assembly will be unheated. The applicant will block the bypass flow paths for this test. The applicant will follow the test plan developed and implemented for the PWR Owners’ Group (PWROG) fuel debris capture testing with regard to debris preparation, the addition of debris, and pressure drop monitoring. The PWROG test plan is consistent with and accounts for revised NRC guidance for the PWRs to respond to GL 2004–02 (ML080230234). The applicant will perform several tests at flow rates ranging from 1 to 5 kg/second (15.9 to 79.3 gpm) and at atmospheric pressure and ambient temperature. The flow rates are representative of the flow at recirculation conditions, and the atmospheric pressures and ambient temperatures result in a viscosity that is conservative with respect to the pressure drop due to debris blockage. The test will be initiated at clean conditions to establish a flow representative of post-LOCA recirculation conditions. The flow will be injected at the fuel assembly inlet. Once a steady-state has been established, the debris will be added to the system in a manner consistent with NRC guidance identified in “NRC Staff Review Guidance Regarding Generic Letter 2004-2 Closure in the Area of Strainer Head Loss and Vortexing” dated March 2008 (ML080230038). The particulate debris will be added first and in such a way that it does not coagulate and will therefore be able to block more of the potential fiber mat interstices. Then the fibrous debris will be slowly added in small amounts, so as to ensure that it does not coagulate but remains as individual fibers. Once all of the particulate and fibrous debris has been added, the chemical surrogate debris will be slowly added in batches so that it does not coagulate. According to the applicant’s description, the particulate debris surrogate is the same debris surrogate that the PWROG fuel debris capture tests used (silicon carbide having a dimension of 0.01 mm (10 microns)), and the chemical surrogate debris will be prepared using the method identified in WCAP-16530-NP-A. The applicant will monitor the pressure drop across the inlet and the entire fuel assembly. In addition, the applicant will monitor the flow rate and coolant temperature. The test will be run until all of the debris has been deposited in the system and a steady-state pressure drop condition has been achieved.

The staff generally agrees with the applicant’s test plan as outlined above and expects that the number of tests will be sufficient to determine the uncertainty in flow rate and differential pressure measurements. The staff expects the applicant to properly evaluate the flow and measurement uncertainties. Because most of the crud will be deposited in the fuel inlet orifice and the first grid, testing of the partial assembly is acceptable. The GSI-191 fuel assembly testing performed by the PWROG to address the downstream effects of debris on fuel used subcooled water. The use of subcooled water provides an adequate test because the acceptance criterion is based on the ratios of initial and final pressure drops and flow rates, which account for the impact of coolant temperature and density. Testing at cold conditions is acceptable. The sequence of the addition of the debris for the test follows the PWROG test protocol, which is acceptable because the staff observed limited testing and accepted the test results in the PWROG test. The staff noted that the applicant has applied the lessons learned from the PWROG tests. The debris type and quantity listed in the table in FSAR Section 6C.3.1.9.3 was previously evaluated in this SER section. The staff noted that the applicant will submit a detailed test protocol including test procedures, as indicated in the license condition for the staff’s review 24 months before fuel loading.

The staff concluded that the applicant’s test plan, analysis approach, analysis conclusions, and debris assumptions for the downstream tests are acceptable because the applicant uses numerous conservative assumptions in the calculation that include the following:

(1) No credit is taken for the HPCF in the analysis to determine the test flow requirement; assuming a single failure, at least one HPCF will be available.

(2) The test flow requirement is based on a constant decay heat assumed at 5 minutes after shutdown. The decay heat decreases exponentially during a post-LOCA core cooling.

(3) No credit is taken for the design fuel assembly bypass flow in the analysis to determine the test flow requirement.

(4) The test acceptance criterion for the allowable pressure drop is reduced by a factor of 4 to provide conservatism. Although the GE 8 x 8 fuel and the Optima-2 fuel designs are different, they are hydraulically similar and any uncertainties related to design differences are accounted for by the k loss increase by the factor of 4.

(5) Test acceptance conditions are based on the hottest fuel assembly. Most of the fuel assemblies are at significantly lower power levels.

(6) The peak cladding temperature is unchanged by the analysis with debris plugging because the peak cladding temperature is reached within seconds of a LOCA initiation, and debris requires minutes to reach the fuel; Since no core is uncovered, an increase in peak cladding temperature is not expected.

(7) Maximum clad temperatures for the hot fuel bundle after debris plugging are less than 250° C (482° F). There is a significant margin from the acceptance criteria specified in 10 CFR 50.46.

The acceptance criteria proposed in the license condition used for the test is acceptable. The staff verified through an audit that the applicant's calculation supports the license condition, which shows that the void fraction of 0.95 is maintained throughout the long-term cooling period. Also, the calculated peak cladding temperature is well within the acceptance criteria specified in 10 CFR 50.46. There are diverse ECCS injection sources and injection paths to the core. The staff found reasonable assurance that there will be adequate core cooling for the long-term period. RAIs 04.04-3 and 04.04-4 are resolved, and Open Item 04.04-3 is therefore closed.

Administrative Departure

- STD DEP Vendor

The applicant replaces "Toshiba," for "GE" in Appendix 6C. 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 found that this administrative departure does not affect the presentation of any design discussion or qualification of the design margin and is therefore acceptable.

The applicant's evaluation in accordance with 10 CFR Part 52, Appendix A, Section VIII, Item B.5 determined that these departures do 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. In addition, the applicant's process for evaluating departures and other changes to the DCD is subject to NRC inspections.

- STD DEP Admin Administrative Departure

The changes in Subsection 6.2.1.1.7 of COL FSAR Section 6.2.1, Sections 3B.5 and 3B.7, and Subsections 3B.2.2.3, 3B.3.3, 3B.4.2.3, 3B.4.3.2.1, and 3B.4.3.3.3.3 are administrative in nature and are therefore acceptable.

The applicant identifies these departures as not requiring prior NRC approval. Within the review scope of this section, the staff found it reasonable that this departure is characterized as not requiring prior NRC approval, per 10 CFR Part 52, Appendix A, Section VIII.B.5.

COL License Information Items

- COL License Information Item 6.4 Suppression Pool Cleanliness

As stated in Subsection 6.2.7.3 of the application, Appendix 6C discusses suppression pool cleanliness to prevent ECCS suction strainer plugging, in accordance with Subsection 6.2.1.7. Subsection 6.2.7.3 of the application also states that "periodic inspections of the suppression pool for cleanliness are performed during outage periods. Maintenance procedures provide procedure steps for removing, at periodic intervals, sediment and floating or sunk debris from the suppression pool that the [suppression pool cleanup unit] does not remove."

The application was not clear about the frequency of periodic inspections of the suppression pool cleanliness and what administrative controls would be available to the applicant for periodic inspections. Therefore, NRC staff issued **RAI 06.02.02-7** requesting the applicant to state the frequency of the periodic inspections of suppression pool cleanliness and include these inspections as TS Surveillance Criteria. The applicant's response to **RAI 06.02.02-7** dated September 28, 2009 (ML093730448), states that suppression pool cleanliness will be an Operational Program that is listed in Section 13.4S and will be implemented before startup testing. The applicant has added FSAR Subsection 6.2.1.7.1 to describe this program. As stated in FSAR Subsection 6.2.1.7.1, the suppression pool cleanup system will normally be operated in alignment with a train of the fuel pool cleanup filter/demineralizers to ensure suppression pool water quality. Floating debris and sediment in the suppression pool that are not removed by the SPCU system will be removed during refueling outages. In the unlikely event of a primary containment entry during the operating cycle, a close-out inspection will be performed before the return to operation. This response addressed the staff's concern on the frequency of periodic inspections of suppression pool cleanliness.

The applicant's response also states that NUREG–1434, "BWR Standard Technical Specifications, General Electric Plants BWR/6," does not include surveillance of suppression pool cleanliness. However, the applicant believes that an Operational Program as described in FSAR Subsection 6.2.1.7.1 is adequate to ensure suppression pool cleanliness. After reviewing the applicant's position and FSAR Subsection 6.2.1.7.1, the staff determined that inspections on suppression pool cleanliness would not warrant a TS Surveillance Criterion. The applicant's response addresses the staff's concerns and is acceptable. Therefore**, RAI 06.02.02-7** is closed.

During the audit (ML092370709), the applicant stated that the plan is to eliminate all of the fiber insulation in the STP Units 3 and 4 containments and to minimize other debris by adopting Institute of Nuclear Power Operators (INPO) and Electric Power Research Institute (EPRI) guidance for cleanliness and foreign material exclusion. The staff issued **RAI 06.02.02-5** requesting the applicant to provide INPO and EPRI guidance and update the FSAR to describe the implementation. In response to this RAI (ML092730448), the applicant agreed to update the FSAR by adding Subsection 6.2.1.7.1 on the Suppression Pool Cleanliness Program. The purpose of this Operational Program is to ensure that the primary containment is free from debris that could become dislodged in an accident and be transported to the ECCS suction strainers and interfere with their proper functioning during a design-basis event. This program applies to the primary containment, including the drywell and suppression pool, for STP Units 3 and 4. The program is comprised of (1) a design change control to ensure that material whose susceptibility to damage resulting in uncontrolled debris is limited and cannot be replaced with material that has a greater susceptibility; (2) restricted access to the primary containment during reactor operation and refueling periods; (3) a suppression pool cleanup system operation to maintain suppression pool cleanliness; (4) foreign material exclusion and housekeeping requirements to ensure that foreign material that could be detrimental to the ECCS strainer operation if left in the primary containment is removed before containment close out; and (5) drywell, suppression pool, and strainer inspection following outages to ensure that no debris is present before the containment is closed in preparation for operation. The staff reviewed the information on the Suppression Pool Cleanliness Program and determined that this program will limit the quantity of latent debris in the STP Units 3 and 4 containments. This information addresses the staff's concerns raised in **RAI 06.02.02-5** and is therefore acceptable. Thus, **RAI 06.02.02-5** is closed.

- COL License Information Item 6.5 Wetwell-to-Drywell Vacuum Breaker Protection

Specific information that the applicant shall provide to address COL License Information Item 6.5 includes appropriate design features with complete structural shielding of vacuum breaker valves from pool swell loads. The structural shielding features should be designed for pool swell loads based on the methodology approved for the Mark II/III designs and for pool swell loads defined to the maximum practical extent.

The applicant proposes to provide a vacuum breaker shield (consisting of a solid "V"- shaped plate) below each vacuum breaker to protect the valves from LOCA pool swell loads. However, the applicant does not provide an actual design of the shield in the COL FSAR, Revision 3.

NRC staff found that the applicant's information addressing this COL license information item is inadequate. The staff issued RAI 06.02.01.01.C-17 requesting the applicant to provide a detailed design of the vacuum breaker shield to resolve and close this issue. This RAI was tracked as an open item in the SER with open items. The applicant's response to **RAI 06.02.01.01.C-17** dated May 26, 2010 (ML101530167), provides a preliminary design of the vacuum breaker shield. The applicant states that the design will be finalized after the completion of the pool swell loads structural evaluation. This response resolved Open Item 06.02.01.01.C-17**.**

6.2.1.5 *Post Combined License Activities*

The applicant identifies the following commitment:

- Commitment (COM 6C-1) – Perform a downstream effects analysis for components (pumps, valves, and heat exchangers) in accordance with WCAP-16406-P and the accompanying SER and submit the evaluation to the NRC 18 months before fuel loading.

The staff proposes that the following license condition be satisfied before fuel can be loaded into the core:

License Condition 06.02-1:

A downstream fuel effects test must be conducted and the results provided to the NRC no later than 18 months before fuel loading. The test plan, analysis basis, and debris assumptions are described in Appendix C, Section 6C.3.1.8. The test procedures will be provided to the NRC no later than 24 months before fuel loading. The acceptance criteria for this test are based on the following equation:

$(\ P_f/\quad {}_i)^2 \leq 1200 \text{ x } (w_f/w_i)^2$

Where:

subscript "i" denotes initial (i.e., unfouled) conditions, and "f" indicates fouled conditions, w is the flow rate into the assembly, and Δp is the pressure drop from the bundle inlet to downstream of the third grid

Initial fuel loading will not be allowed until this condition is satisfied.

6.2.1.6 *Conclusion*

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

The staff's review found that the applicant is in compliance with the relevant NRC regulations, including the acceptance criteria outlined in NUREG–0800 Section 6.2.1 and other NRC RGs. The applicant has adequately addressed COL License Information Items 6.4 and 6.5.

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

The staff concluded that STP DEP 6.C-1 conforms to the guidance in RG 1.82, Revision 3 and Topical Report NEDO-32686-A. This departure therefore complies with 10 CFR 50.46(b)(5), as it relates to debris protection for ECCS strainers.

6.2.2 Containment Heat Removal System

Section 6.2.2 of the STP Units 3 and 4 COL FSAR incorporates by reference, with no departures or supplements, Section 6.2.2, "Containment Heat Removal System," of the ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding issue related to this 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 containment heat removal system have been resolved.

6.2.3 Secondary Containment Functional Design

Section 6.2.3 of the STP Units 3 and 4 COL FSAR incorporates by reference, with no departures or supplements, Section 6.2.3, "Secondary Containment Functional Design," of ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[2] The staff's review confirmed that there is no outstanding issue related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the secondary containment functional design have been resolved.

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

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

6.2.4 Containment Isolation System

6.2.4.1 *Introduction*

This section of the FSAR addresses the isolation systems, including valves and associated piping, which allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products from postulated accidents.

6.2.4.2 *Summary of Application*

Section 6.2.4 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 6.2.4 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, the applicant provides the following:

Tier 1 Departure

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

This departure addresses the design changes to the instrumentation and control (I&C) architecture to accomplish the following:

- Elimination of obsolete data communication technology.
- Elimination of unnecessary, inadvertent actuation prevention logic and equipment.
- Clarifications of digital controls nomenclature and systems.
- Final selection of platforms that change the implementation architecture.
- Testing and surveillance changes.

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 6.2-3 Containment Penetrations and Isolation

This standard departure corrects primary containment penetration errors and inconsistencies in Section 6.2 of the referenced ABWR DCD and provides an additional design detail that was not in the referenced ABWR DCD. This departure is the result of a detailed 3-D layout analysis that was performed to ensure that the penetrations meet U.S. codes and standards for mechanical and electrical separation. Changes to the tables include corrections of the containment penetration elevation, azimuth, offset, and diameter.

6.2.4.3 *Regulatory Basis*

The regulatory basis of the information incorporated by reference is in NUREG–1503. In addition, the relevant requirements of the Commission's regulations for the containment isolation system, and the associated acceptance criteria, are in Section 6.2.4 of NUREG–0800.

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

6.2.4.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.2.4 of the certified ABWR DCD. The staff reviewed Section 6.2.4 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

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

This departure is evaluated in Chapter 7 of this SER.

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 6.2-3 Containment Penetrations and Isolation

This departure is the result of a detailed 3-D layout analysis that was performed to ensure that the ABWR design (1) conforms to U.S. codes and standards, (2) corrects errors and inconsistencies in the referenced ABWR DCD, (3) revises penetration locations to ensure that they meet separation criteria based on the 3-D layout analysis, and (4) provides additional design information regarding containment isolation valve testing that was not in the referenced ABWR DCD.

This departure affects the detailed containment isolation valve listings in Tables 6.2-5, 6.2-6, 6.2-7, 6.2-8, and 6.2-10. These tables also include changes due to containment isolation aspects of the breathing air system discussed in Subsection 9.3.7.6 (STP DEP 9.3-2). According to the applicant, these changes (1) collectively ensure that the design fully complies with NRC rules and regulations, (2) do not impact the probability of the occurrence of accidents or the consequence of accidents, and (3) do not create accidents of a different type that have not been evaluated. These changes do not adversely affect the containment fission product barrier. According to the applicant, there is no change in any method of analysis and no adverse effect on severe accident mitigation.

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

6.2.4.5 *Post Combined License Activities*

There are no post COL activities related to this section.

6.2.4.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG-1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the relevant 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 containment isolation system that were incorporated by reference have been resolved.

The staff found it reasonable that the identified Tier 2 departures are adequately characterized as not requiring prior NRC approval. The staff concluded that the applicant has provided sufficient information to satisfy Section 6.2.4 of NUREG–0800.

6.2.5 Combustible Gas Control In Containment

6.2.5.1 *Introduction*

This section of the FSAR addresses the systems designed to monitor and control hydrogen and oxygen gas in the containment. The objective of the atmospheric control systems is to maintain an inert atmosphere inside the primary containment during all plant operating modes, except during shutdown periods for refueling or equipment maintenance and during limited periods of time to permit access for inspections at low reactor power. Following a LOCA, if a sufficient amount of combustible gas is generated inside the containment, the gas may react with the oxygen in the containment at a rate rapid enough to breach the containment or cause a leakage rate that exceeds TS limits. Additionally, the associated pressure and temperature increases could damage systems and components essential to the continued control of post-accident conditions.

6.2.5.2 *Summary of Application*

Section 6.2.5 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 6.2.5 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, the applicant provides the following:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This standard departure eliminates the hydrogen recombiner requirements of the certified ABWR design.

COL License Information Item

- COL License Information Item 6.2 Alternate Hydrogen Control

This COL license information item addresses the revision of 10 CFR 50.44, which amended the standards for combustible gas control in light-water-cooled power reactors by eliminating the requirements for hydrogen recombiners and relaxing the requirements for hydrogen and oxygen monitoring.

6.2.5.3 *Regulatory Basis*

The regulatory basis for the information incorporated by reference is in NUREG–1503. In addition, the relevant requirements of the Commission regulations for the combustible gas control in the containment, and the associated acceptance criteria, are in Section 6.2.5 of NUREG-0800.

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

The regulatory basis for reviewing the COL license information items is in Section 6.2.5 of NUREG–0800.

6.2.5.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.2.5 of the certified ABWR DCD. The staff reviewed Section 6.2.5 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

The staff reviewed the information in the COL FSAR:

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

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This departure is based on 10 CFR 50.44, “Combustible gas control for nuclear power reactors,” which was amended after the issuance of the design certification for the ABWR. The departure reflects the elimination of the requirement to maintain the equipment needed to mitigate a design-basis LOCA hydrogen release. The departure eliminates the ABWR FCS, which consists of two redundant hydrogen recombiners and is no longer required in the response to a design-basis LOCA. Also, the containment hydrogen and oxygen monitoring instruments are no longer classified as Category 1. The ACS establishes and maintains the containment atmosphere to less than 3.5 percent by volume oxygen during normal operating conditions to maintain an inert atmosphere.

NRC staff reviewed the proposed standard departure in COL application Part 7, “Departures Report,” Section 2.0, with respect to the Commission rules and regulations. The applicant’s evaluation of this departure shows that the design complies with the revisions to the regulation

[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.

for controlling combustible gases added after the issuance of the design certification for the ABWR. The proposed elimination of the hydrogen recombiner requirements of the certified ABWR design is in accordance with 10 CFR 50.44, which was amended after the issuance of the design certification for the ABWR. Because this is a standard departure applicable to all COL applicants referencing the ABWR DCD, no loss of standardization will result from the departure. The staff determined that this standard departure is consistent with Commission rules and regulations and is therefore acceptable.

COL License Information Item

- COL License Information Item 6.2 Alternate Hydrogen Control

10 CFR 50.44 amended standards for combustible gas control in light-water-cooled power reactors. This amended rule eliminates the requirements for hydrogen recombiners and relaxes the requirements for hydrogen and oxygen monitoring. With the elimination of the requirement to provide hydrogen control equipment, the need to provide a cost analysis for an alternate control system is also eliminated. The staff reviewed COL License Information Item 6.2, as discussed in FSAR Subsection 6.2.7.1 for inerted containments, and found it acceptable because it complies with 10 CFR 50.44.

6.2.5.5 *Post Combined License Activities*

There are no post COL activities related to this section.

6.2.5.6 *Conclusion*

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

The staff reviewed the proposed standard departure with respect to Commission rules and regulations, per Section VIII.A.4 of Appendix A to 10 CFR Part 52. The staff determined that the standard departure is consistent with Commission rules and regulations and has no adverse impact on public health and safety.

The staff found that the applicant has adequately addressed COL License Information Item 6.2, which can be considered closed. In conclusion, the applicant has provided sufficient information to satisfy Section 6.2.5 of NUREG–0800.

6.2.6 Containment Leakage Testing

6.2.6.1 *Introduction*

This FSAR section addresses the Leakage Rate Testing Program for the reactor containment. Testing requirements assure that the containment's leak-tight integrity can be verified throughout the lifetime of use. Additionally, periodic Type A, B, and C testing must be performed to assure that leakage through the containment systems and components that penetrate the primary containment does not exceed the allowable leakage rate values specified in the standard TS.

6.2.6.2 *Summary of Application*

Section 6.2.6 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 6.2.6 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, the applicant provides the following:

COL License Information Items

- COL License Information Item 6.3 — Administrative Controls Maintaining Containment Isolation

This COL license information item addresses the controls for maintaining the primary containment boundary located in various plant operating procedures that control the operation, testing, and maintenance requirements for the containment barriers.

- COL License Information Item 6.5a — Containment Penetration Leakage Rate Test (Type B)

This COL license information item states that "Type B leakage rate tests are performed in conformance with 10 CFR Part 50, Appendix J for containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds, airlocks and lock door seals, equipment and access hatch seals, and electrical canisters, and other such penetrations."

Supplemental Information

- **Operational Program 7** — Containment Leakage Rate Testing Program

The applicant provides this information in FSAR Section 13.4S, Table 13.4S-1, "Operational Programs Implementation Required by NRC Regulation."

6.2.6.3 *Regulatory Basis*

The regulatory basis for the review of the information incorporated by reference is in NUREG-1503. In addition, the relevant requirements of the Commission regulations for the containment leakage testing, and the associated acceptance criteria, are in Section 6.2.6 of NUREG–0800.

COL License Information Items 6.3 and 6.5a are satisfied based on meeting the requirements of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

6.2.6.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.2.6 of the certified ABWR DCD. The staff reviewed Section 6.2.6 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

The staff reviewed COL License Information Items 6.3 and 6.5a and Operational Program 7.

COL License Information Items

- COL License Information Item 6.3 — Administrative Control Maintaining Containment Isolation

Specific information provided by the applicant to address COL License Information Item 6.3 includes maintaining the primary containment boundary by administrative controls, in accordance with DCD Subsection 6.2.6.3.1. These controls are in various plant operating procedures to control access, surveillance, and maintenance for testing and restoring containment components and for controlling the routine operation of containment valves and components. The staff found that the applicant has adequately addressed COL License Information Item 6.3 for the primary containment valves and components (Type C), in accordance with requirements of 10 CFR Part 50, Appendix J.

- COL License Information Item 6.5a — Containment Penetration Leakage Rate Test (Type B)

Specific information provided by the applicant to address COL License Information Item 6.5a includes performing Type B leakage rate tests in conformance with 10 CFR Part 50, Appendix J, for containment penetrations whose designs incorporate resilient seals, bellows, gaskets or sealant compounds, airlocks and lock door seals, equipment and access hatch seals, electrical canisters, and other similar penetrations described in DCD Subsection 6.2.6.2.1. NRC staff found that the applicant has adequately addressed COL License Information Item 6.5a for primary containment penetration seals, bellows, gaskets, airlock and hatch seals (Type B), in accordance with requirements of 10 CFR Part 50, Appendix J.

Supplemental Information

- Operational Program 7 — Containment Leakage Rate Testing Program

The applicant provides implementation schedules and milestones to address the relevant operational program associated with the containment leakage rate testing program in FSAR Section 13.4S.

NRC staff reviewed the applicant's proposal using the review procedures described in Section 6.2.6 of NUREG–0800, Revision 3. The applicant indicates that the procedures will be developed before fuel loading. The staff found the schedule for the development of the procedures as well as their scope to be reasonable. The staff found the information in COL FSAR Section 13.4S, "Operational Program Implementation of the Containment Leak Rate Testing Program," acceptable based on the requirements of 10 CFR Part 50, Appendix J.

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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.

6.2.6.5 *Post Combined License Activities*

There are no post COL activities related to this section.

6.2.6.6 *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 containment leakage testing that were incorporated by reference have been resolved.

The staff compared the information in the application to the relevant NRC regulations, the acceptance criteria defined in NUREG–0800, Section 6.2.6, and in other NRC regulatory guides. The staff concluded that the applicant is in compliance with NRC regulations. The applicant has adequately addressed COL License Information Items 6.3 and 6.5a, which are considered closed. The applicant has adequately addressed the Operational Program 7 regarding the containment leakage rate testing, which can be considered closed. In conclusion, the applicant has provided sufficient information to satisfy Section 6.2.6 of NUREG–0800, and no outstanding information is expected to be addressed in the COL FSAR related to this section.

6.3 <u>Emergency Core Cooling Systems</u>

6.3.1 Introduction

The design-basis and performance evaluation of the ECCS for the STP Units 3 and 4 ABWR is in FSAR Section 6.3.

The ECCS proposed for STP Units 3 and 4 consist of the following:

- RCIC system
- HPCF system
- ADS
- LPFL mode of the RHR system

The RCIC system injects water into a feedwater line using a pump driven by a steam turbine. The HPCF system is comprised of two independent loops providing the emergency makeup water to the RPV. If the RCIC and HPCF systems cannot maintain the RPV water level, the ADS, which consists of pressure relief valves, reduces the RPV pressure so that the flow from the RHR system operating in an LPCF mode can be injected into the RPV.

6.3.2 Summary of Application

Section 6.3 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 6.3 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. As required by Section IV.A.3 of the ABWR Design Certification Rule, the applicant also provides the proprietary information referenced in the ABWR DCD.

In addition, in FSAR Section 6.3, the applicant provides the following:

Tier 1 Departures

- STD DEP T1 2.4-1 RHR System and Spent Fuel Cooling

This departure adds the capability for a third RHR loop, RHR Division A, in the augmented FPC and fuel pool makeup modes.

- STD DEP T1 2.4-3 RCIC Turbine/Pump

The departure replaces the RCIC turbine and pump system design with an integrated (monoblock) alternate turbine/pump system design.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP Admin Administrative Departure

The applicant incorporates by reference Appendix 6D of the certified ABWR referenced in 10 CFR Part 52, Appendix A. The following administrative departure corrects an editorial error in the confirmation equation for the low-pressure injection flow portion of the HPCF:

$$P_{727} = H_{727} + H_s + \cancel{70.68}\ 0.69\ MP_a + \text{margin}$$

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

This departure uses expanded graphite packing to seal the valve stem penetration of large bore reactor coolant pressure boundary isolation valves. Due to the reduced valve stem leakage, the valve stem leak-off lines, this departure eliminates the valve gland leak-off lines and related leak detection instrumentation.

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

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

In Tables 6.3-8 and 6.3-9, the required NPSH values for the HPCF and RHR pumps were changed from 2.2 m to 1.7 m and 2.4 m to 2.0 m, respectively.

COL License Information Items

- COL License Information Item 6.6 ECCS Performance Results

The applicant commits (COM 6.3-1) to provide the following as an amendment to the FSAR and in accordance with 10 CFR 50.71(e), at least 1 year before fuel loading:

> The exposure-dependant maximum average planar linear heat generation rate (MAPLHGR), peak cladding temperature, and oxidation fraction for each initial

core bundle design based on the limiting break size. The analysis will reflect the final fuel design for the initial core loading.

- COL License Information Item 6.7 ECCS Testing Requirements

The applicant commits (COM 6.3-2) to perform the ECCS testing in accordance with the TS during every refueling outage in which each subsystem of the ECCS is actuated through the emergency operating sequence. The applicant will develop the test procedure consistent with the plant operating procedure development plan in Section 13.5.

- COL License Information Item 6.7a Limiting Break Results

The applicant commits (COM 6.3-3) to provide as an amendment to the FSAR and in accordance with 10 CFR 50.71(e) at least 1 year before fuel loading, the results of the analysis on the limiting break for each bundle design. The applicant adds that the analysis will reflect the final fuel design for the initial core loading.

6.3.3 Regulatory Basis

The regulatory basis for the information incorporated by reference is in NUREG–1503. In addition, the relevant requirements of the Commission regulations for the ECCS, and the associated acceptance criteria, are in Section 6.3 of NUREG–0800.

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

The review and acceptability of COL License Information Items 6.6, 6.7, and 6.7a are based on meeting the applicable acceptance criteria in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors"; Appendix A to 10 CFR Part 50, GDC 37, “Testing of emergency core cooling system”; Appendix K, “ECCS Evaluation Models,” performance requirements; and the guidance of RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants.”

6.3.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.3 of the certified ABWR DCD. The staff reviewed Section 6.3 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and the information incorporated by reference address the relevant information related to this section.

[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 Departures

- STD DEP T1 2.4-1 RHR System and Spent Fuel Cooling

The ABWR DCD has two RHR loops in the augmented FPC and fuel pool makeup modes. This departure adds a third RHR loop, RHR Division A, in the augmented FPC and fuel pool makeup modes, which provides the ability to supply spent FPC or makeup from any of the three RHR loops.

This departure is evaluated in Section 5.4.7 of this SER. An evaluation of the changes to the ITAAC associated with this departure is in Section 14.3 of this SER.

- STD DEP T1 2.4-3 RCIC Turbine/Pump

The applicant has changed the design of the RCIC turbine and pump assembly in favor of an improved design. The new RCIC turbine/pump is a monoblock design consisting of a horizontal, two-stage centrifugal water pump driven by a steam turbine contained in a turbine casing integral with the pump casing. This improved design offers system simplification due to (a) the monoblock design that places the pump and turbine within the same casing, (b) a shaft seal is not required, (c) a barometric condenser is not required, (d) an oil lubrication or oil cooling system is not required because the system is totally water lubricated, (e) a steam bypass line is not required for startup, (f) the simpler auxiliary subsystems, and (g) a vacuum pump and associated penetration piping or isolation valves are not required.

This departure is evaluated in Section 5.4.6 of this SER. Section 14.3 of this SER evaluates the changes to the ITAAC associated with this departure.

Tier 2 Departures Not Requiring Prior NRC Approval

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

This departure is reviewed in SER Section 7.3.

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

This departure is reviewed in SER Section 6.2.

- 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 an administrative departure in Subsections 6.3.3.2 and 6.3.3.10. NRC staff found that these administrative departures do not affect the presentation of any design discussion or the qualification of a design margin and is therefore acceptable.

The applicant's evaluation in accordance with 10 CFR Part 52, Appendix A, Section VIII, item B.5 determined that these departures do 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. In addition, the applicant's process for evaluating departures and other changes to the DCD is subject to NRC inspections.

FSAR Section 6D.2.4, "Low Pressure Injection Flow"

The applicant provides an administrative change to the P_{727} equation in FSAR Section 6D.2.4 as a typographical correction of the value from "70.68" to "0.69." NRC staff reviewed DCD Tier 2 Figure 6D-1, "Injection Flow," to determine whether the figure is consistent with the proposed change. The staff found that H_{182} and H_{727} are not clearly defined. The staff issued RAI 06.03-1 (eRAI 2470) requesting the applicant to clarify the definitions of H_{182} and H_{727}, including their applicability to DCD Tier 2 Figure 6D-1 and Tables 6.3-1 and 6.3-8.

The applicant's response to **RAI 06.03-1** dated May 13, 2009,(ML091380169), provides the technical bases for the change. The applicant states that according to the following equation, the pressure head across the pump for the low-pressure flow of 727 m^3 is

$$P_{727} = H_{727} + H_s + 0.69\ MP_a + \text{margin}$$

Where:

H_{727} – Hydraulic head loss for the flooder line, which is a function of the square of the flow rate

H_s – Static head, which is equal to the difference in fluid elevation between the RPV and the suppression pool water level

$0.69MP_a$ is the differential or gauge pressure between the RPV and the suppression pool airspace

In DCD Tier 2 Figure 6D-1, the HPCF flow rate is specified as 727 m^3/hour (h) at a differential pressure of 0.69 MP_a. DCD Tier 2 Table 6.3-8 specifies this flow as 0.79 MP_a, which is equivalent to a differential or gauge pressure of $0.69MP_a$. The correct value for the third term in the above equation is "0.69 MP_a," not "$70.68MP_a$." This change makes the equation consistent with other sections of the DCD. The staff found it reasonable that this departure does not require prior NRC approval. The staff found the applicant's response to this RAI acceptable, and RAI 06.03-1 is closed.

<u>FSAR Tables 6.3-8 and 6.3-9</u>

In DCD Tables 6.3-8 and 6.3-9, HPCF and RHR pumps NPSH required were changed from 2.2 m to 1.7 m and 2.4 m to 2.0 m, respectively. This change is reviewed in Section 6.2.

<u>COL License Information Item</u>

- COL License Information Item 6.6 ECCS Performance Results

The current licensing basis for STP Units 3 and 4 includes the fuel design described in the DCD, which is incorporated by reference. Consequently, the exposure-dependent MAPLHGR, the peak cladding temperature, and the oxidation fraction for the core bundle design based on the limiting break size are consistent with ABWR DCD Section 6.3.3. Hence, COL License Information Item 6.7 is satisfied.

- COL License Information Item 6.7 ECCS Testing Requirements

The applicant has committed (COM 6.3-2) to perform the ECCS testing during every refueling outage in which each ECCS subsystem is actuated through the emergency operating sequence. A reference to NEDO–33297, "ABWR Procedure Development Plan," dated January 2007 was deleted. The applicant will develop the test procedure to be consistent with the plant operating procedure development plan in Section 13.5. The staff found this commitment acceptable.

- COL License Information Item 6.7a Limiting Break Results

The current licensing basis for the STP Units 3 and 4 includes the fuel design described in the DCD, which is incorporated by reference. Consequently, the analysis for the limiting break for the core bundle design is in Subsection 6.3.7.3 of the DCD. Hence, COL License Information Item 6.7a is satisfied.

6.3.5 Post Combined License Activities

The applicant identifies the following commitment:

- Commitment (COM 6.3-2) – Perform the ECCS testing during every refueling outage in which each ECCS subsystem is actuated through the emergency operating sequence, in accordance with the TS described in ABWR DCD Subsection 6.3.4.1. Also, develop the test procedure consistent with the plant operating procedure development plan in Section 13.5.

6.3.6 Conclusion

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

The staff compared the information in the application to the relevant NRC regulations, the acceptance criteria defined in NUREG–0800, Section 6.3, and other NRC regulatory guides. The staff found it reasonable that the identified Tier 2 departures are characterized as not requiring prior NRC approval, per 10 CFR Part 52, Appendix A, Section VIII.B.5. The staff concluded that the applicant is in compliance with NRC regulations. The applicant has adequately addressed COL License Information Items 6.6, 6.7, and 6.7a, which are considered closed.

6.4 Habitability Systems

6.4.1 Introduction

This section of the FSAR addresses the control room habitability system that will provide (1) missile protection, (2) radiation shielding, (3) radiation monitoring, (4) air filtration and ventilation, (5) lighting, (6) personnel and administrative support, and (7) fire protection. The control building HVAC system will maintain the control room ambient temperature at a habitable level to permit prolonged personnel occupancy throughout a postulated DBA. The system's design provides control room pressurization, with respect to the surrounding spaces and filtered intake during accident situations and for purging smoke and toxic gases. The system is capable of automatically transferring from its normal operating mode to its emergency or isolation modes upon detection of adverse conditions (e.g., high radiation or smoke). Backup power sources are provided for the essential components of the HVAC system. Section 9.4.1 of the FSAR provides more information on the control building HVAC system. The habitability systems are designed to detect and limit the introduction of radioactive material and smoke into the control room. The NRC staff's review of the control room radiological habitability is in Section 15.6, "Decrease in Reactor Coolant Inventory," of this SER.

6.4.2 Summary of Application

Section 6.4 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 6.4 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 6.4, the applicant provides the following:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 9.4-2 Control Building HVAC System

This standard departure changes the smoke removal mode of operation of the control building HVAC system, as described in FSAR Subsections 6.4.4.2 ("Smoke and Toxic Gas Protection"), 9.4.1.1.4 ("Safety Evaluation"), and 9.5.1.1.6 ("Smoke Control System"). The departure adds a main air supply duct bypass line around the control room air handling unit (AHU). When the recirculation damper is closed and the damper in the bypass duct around the AHU is opened, the air exhaust and supply are balanced and the smoke is exhausted rather than transported to other areas of the control building.

COL License Information Item

- COL License Information Item 6.8 Toxic Gases

GDC 19 is related to providing adequate protection to permit access and occupancy of the main control area envelope under accident conditions. Acceptance is based on conforming with the guidance of RG 1.78 relating to instrumentation to detect and alarm any hazardous chemical release in the plant vicinity and to the capability of the system to isolate the main control area envelope from such releases. Acceptance is also based on conforming with the guidance of RG 1.78 relating to the capability of the system to limit the accumulation of chlorine within the main control area envelope. The ABWR is not designed for any particular hazardous chemical release except for exterior smoke. The main control area envelope is provided with a filtration system for releasing radioactivity and can be easily modified for isolation signals to handle additional toxic chemical sensors. According to ABWR DCD Subsection 6.4.4.2, chemical accidents (including chlorine) require site-specific information such as frequency, distance from the control room, and size of the container.

In Section 6.4.7 of the FSAR, the applicant provides the following site-specific supplement to address COL License Information Item 6.8:

Based on analyses in Section 2.2S.3, no hazardous chemicals with quantities exceeding the criteria of RG 1.78 have been identified. Therefore, instrumentation to detect and alarm a hazardous chemical release in the STP Units 3 and 4 vicinity and to isolate the main control area envelope from such releases is not required.

6.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is in NUREG–1503. In addition, the relevant requirements of the Commission regulations for the habitability systems, and the associated acceptance criteria, are in Section 6.4 of NUREG–0800.

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

In addition, the relevant requirements of NRC regulations for COL License Information Item 6.8, and the associated acceptance criteria, are in Section 6.4 of NUREG–0800 and RG 1.78.

6.4.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.4 of the certified ABWR DCD. The staff reviewed Section 6.4 of the STP Units 3 and 4 COL FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and the information incorporated by reference address the relevant information related to this section.

The staff reviewed the information in the COL FSAR:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 9.4-2 Control Building HVAC System

This departure adds a main air supply duct bypass line around the control room AHU. This design change facilitates smoke removal in the operation of the control building HVAC system.

[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 applicant's evaluation in accordance with Section VIII.B.5 of Appendix A 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 applicant's process for evaluating departures and other changes to the DCD is subject to NRC inspections.

COL License Information Item

- COL License Information Item 6.8 Toxic Gases

The applicant identifies no hazardous chemicals in Section 2.2S.3 with quantities exceeding the screening criteria of RG 1.78. As a result, the applicant does not provide instrumentation to detect and alarm a hazardous chemical release in the STP Units 3 and 4 vicinity and to isolate the service building clean area from such releases.

The staff's review of the applicant's screening of hazardous material showed that the site-specific supplemental information in the COL application does not address the toxic gas evaluation for control room habitability. NRC staff cannot determine whether the applicant's findings regarding toxic gas evaluation are reasonable. The staff's confirmatory analysis of the STP Units 3 and 4 control room habitability showed that among the hazardous chemicals listed in Tables 2.2S of the FSAR, the following chemicals pose toxic gas threats to the control room: acetic acid (water transport); gasoline (water transport); sodium hypochlorite (onsite storage); 1-hexene (offsite storage); and acetic acid (offsite storage). The staff's computer simulations showed that an accidental release of these hazardous chemicals into the atmosphere would lead to a gas concentration level exceeding the immediate danger to life and health (IDLH) values inside the control room after the resulting gas cloud reaches the control room intake. In this regard, the staff issued **RAI 06.04-1** requesting the applicant to clarify these toxic gas threat evaluations. The staff also requested the applicant to provide details of the toxic gas evaluations, including supporting calculations to demonstrate that these chemical sources are not a threat to the STP Units 3 and 4 control room habitability.

In response to **RAI 06.04-1** (dated September 30, 2009 [ML092750410], and October 29, 2009 [ML093430303]), the applicant submitted the results of the required sensitivity analyses for the toxic gas concentrations inside the control room. The analyses were performed using the ALOHA computer code, and the results showed that the toxic gas concentrations from potentially hazardous chemicals stayed below their IDLH values for a range of atmospheric conditions. However, the staff performed its own confirmatory calculations using the HABIT computer code in accordance with RG 1.78. The staff found that the toxic gas concentrations inside the control room from some of the chemicals significantly exceeded their IDLH limits. The staff's review of the applicant's response also raised questions about the capability of the ALOHA code to analyze liquid spills of large quantities of chemicals, such as the ones involved along the water transportation routes and the offsite storage facilities around STP Units 3 and 4. The staff issued **RAI 06.04-2** requesting the applicant to describe the ALOHA methodology the applicant used and to reconcile the differences between the staff's and the applicant's analyses. This RAI was tracked as open item 06.04-2 in the SER.

The applicant responded to this RAI in a letter dated March 17, 2010 (ML100770387). The staff reviewed the applicant's response to RAI 06.04-2 in conjunction with the information presented during the May 6, 2010 audit of STP Units 3 and 4 Section 6.4, "Toxic Gas Review." As a result of this review, the staff issued RAI 06.04-3 asking for the justifications for a maximum puddle radius of 100 m for acetic acid (offsite storage) and the 1-hour toxic gas simulation cut-off used in ALOHA; and the sensitivity of the chlorine release from sodium hypochlorite (onsite storage) to the ambient temperature. The applicant's response to RAI 06.04-3 dated June 17, 2010 (ML101720634), provides the needed information on the immediate and emergency containment areas and spillway that can confine any acetic acid spill within a 100-m radius near the storage tank. The applicant also provides literature citations substantiating that the 1-hour toxic gas analysis will be conservative. The applicant's calculations show that the chlorine release will not pose a toxic gas threat to the control room, even at zero percent temperature exceedence for the site. The staff concluded that no hazardous chemicals with quantities exceeding the IDLH criteria of RG 1.78 were identified, and there is no toxic gas threat to the STP Units 3 and 4 control room. Open Item 06.04-2 is therefore closed.

In **RAI 06.04-1**, the staff also requested additional information on the thermophysical properties of sodium hypochlorite and gasoline and on the IDLH value of sodium hypochlorite. The staff found the applicant's response to this RAI acceptable regarding the missing thermophysical properties of sodium hypochlorite and gasoline.

6.4.5 Post Combined License Activities

There are no post COL activities related to this section.

6.4.6 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 relevant information relating to the control building HVAC. No outstanding information is expected to be addressed in the COL FSAR related to this subsection. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the control building HVAC system that were incorporated by reference have been resolved.

The staff found it reasonable that Departure STD DEP 9.4.2 is adequately characterized as not requiring prior NRC approval, per 10 CFR Part 52, Appendix A, Section VIII.B.5.

NRC staff's review of the applicant's screening of hazardous material in Section 2.2 showed that the site-specific supplemental information in the COL application has addressed the toxic gas evaluation for control room habitability.

6.5 <u>Fission Products Removal and Control Systems</u>

6.5.1 Engineered Safety Features Filter Systems

6.5.1.1 *Introduction*

This section of the FSAR addresses the ESF filter systems designed to remove fission products from the atmosphere following an accident. The ESF filter systems of the ABWR include the standby gas treatment system (SGTS) and the control room habitability area (CRHA) portion of the HVAC system. The SGTS filters the gaseous effluent from the primary or secondary containment to limit the discharge of radioactivity into the environment. The CRHA portion of the HVAC system is designed to limit the introduction of airborne radioactive materials in the main control area envelope (MCAE).

6.5.1.2 *Summary of Application*

Section 6.5.1 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 6.5.1 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A, with no departures.

In addition, the applicant provides the following:

COL License Information Items

- COL License Information Item 6.9 SGTS Performance

The applicant commits (COM 6.5-1) to provide a secondary containment drawdown analysis in accordance with the ABWR DCD FSER (NUREG–1503), before preoperational testing. The analysis will be based on actual as-built secondary containment and the SGTS design to demonstrate the capability of the SGTS to achieve and maintain the design-negative pressure of 0.25 inch-water gauge (in-wg), within 20 minutes from the time that the secondary containment isolation is initiated following a LOCA.

- COL License Information Item 6.9a SGTS Exceeding 90 Hours of Operation per Year

The applicant states the intent to qualify by analysis the capability of the SGTS system to perform its intended function in the event of a LOCA, if more than 90 hours of operation per year (excluding tests) for either train are anticipated by plant operations based on operation experience.

Appendix 6B

Appendix 6B of the STP Units 3 and 4 COL FSAR incorporates by reference Appendix 6B of the ABWR DCD—including all subsections, tables, and figures—with the following departure:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 7.1-1 References to Setpoints and Allowable Values

This departure changes the terminology from “setpoint” to “nominal setting” for the SGTS adsorber temperature alarm. The setpoint setting for this alarm signal is 155 °C.

6.5.1.3 *Regulatory Basis*

The regulatory basis for the review of the information incorporated by reference is in NUREG-1503. In addition, the relevant requirements of the Commission regulations for the EGF filter systems, and the associated acceptance criteria, are in Section 6.5.1 of NUREG-0800.

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

The regulatory basis for reviewing the COL license information items is in Section 6.5.1 of NUREG–0800. COL License Information Items 6.9 and 6.9a are satisfied based on (1) meeting the requirements of 10 CFR 100, “Reactor Site Criteria”; (2) meeting all applicable requirements of 10 CFR Part 50; and (3) conforming with the provisions of RG 1.52 Section C, except for the revisions to ANSI N509 and ANSI/American Society of Mechanical Engineers (ASME) AG-1 used for the ABWR ESF filter train design. In addition, the acceptance criteria are based on meeting the relevant requirements of Appendix A to 10 CFR Part 50, GDC 41, “Containment atmosphere cleanup”; GDC 42, “Inspection of containment atmosphere cleanup”; and GDC 43, “Testing of containment atmosphere cleanup systems.”

6.5.1.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.5.1 of the certified ABWR DCD. The staff reviewed Section 6.5.1of the STP Units 3 and 4 COL FSAR and considered the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff’s review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

In addition, the staff reviewed the information in the COL FSAR:

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

- COL License Information Item 6.9 SGTS Performance

NRC staff issued RAI 06.05.01-1 requesting the applicant to clarify that COL License Information Item 6.9, “SGTS Performance,” does not eliminate Technical Specification Surveillance Requirement (SR) 3.6.4.1.4 to demonstrate the SGTS capability by performing a physical drawdown test of the reactor building. The applicant’s response to RAI 06.05.01-1 dated July 27, 2009 (ML092050078),clarifies that this analysis will not eliminate the physical drawdown testing required by TS SR 3.6.4.1.4. The staff found the applicant’s response to this RAI addressing this COL license information item acceptable and consistent with 10 CFR 52.79 and 10 CFR 50.36, with regard to TS surveillance requirements. Therefore, RAI 06.05.01-1 is closed.

- COL License Information Item 6.9a SGTS Exceeding 90 Hours of Operation per Year

NRC staff issued **RAI 06.05.01-2** requesting the applicant to provide additional information on COL License Information Item 6.9a, “SGTS Exceeding 90 Hours of Operation per Year.” The staff asked the applicant to demonstrate how an analysis can qualify the charcoal adsorber to the efficiency levels assumed in the design-basis LOCA analysis or how the applicant can perform appropriate laboratory testing of the charcoal adsorber. The applicant’s response to **RAI 06.05.01-2** dated July 22, 2009 (ML092050078) states that laboratory testing of a representative carbon adsorber sample will be performed if the SGTS exceeds 90 hours of operation per year, in accordance with RG 1.52 Revision 2 and TS 5.5.2.7. The staff found the applicant’s response to the RAI addressing this COL information item acceptable and consistent

[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.

with regulatory guidance for assuring filtration system efficacy and filter testing. In the response to **RAI 06.05.01**, the applicant adds that FSAR Subsection 6.5.5.2 will be revised to clarify that the ability of the SGTS to perform its intended function will be demonstrated by appropriate laboratory testing. This RAI was tracked as **Confirmatory Item 6.5-1** in the SER with open items. The staff verified that the applicant has appropriately revised Revision 4 of STP Units 3 and 4 COL Subsection 6.5.5.2, and **Confirmatory Item 6.5-1** is resolved.

In addition, the staff reviewed the relevant information in Appendix 6B of the COL FSAR:

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 7.1-1 References to Setpoints and Allowable Values

The applicant's evaluation of this departure with regard to this section, in accordance with Item B.5 of Section VIII.B.5 of Appendix A, 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 applicant's process for evaluating departures and other changes to the DCD is subject to NRC inspections.

6.5.1.5 *Post Combined License Activities*

The applicant identifies the following commitment:

- COM 6.5-1 – Perform a secondary containment drawdown analysis, in accordance with NUREG–1503 (Page 6-51) before preoperational testing. The applicant will update the FSAR to document the results of the analysis. The applicant clarified in the response to RAI 06.05.0-1 that the analysis would not eliminate physical drawdown testing required by TS SR 3.6.4.1.4.

6.5.1.6 *Conclusion*

The NRC staff's finding related to information incorporated by reference is in NUREG–1503. NRC staff reviewed the application and checked the referenced DCD. The staff's review confirmed that the applicant has addressed the required information relating to the fission products removal and control systems. 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 ESF filter systems that were incorporated by reference have been resolved.

The staff compared the information in the application to the relevant NRC regulations, the acceptance criteria in NUREG–0800 Section 6.5.1, and other NRC RGs. The staff found the applicant's responses to the RAIs that addressed the two COL information items acceptable and consistent with NRC regulations. In addition, the applicant identified STD DEP 7.1-1 in Appendix B to Chapter 6. With respect to the impact of this departure on this section, the staff found it reasonable that this departure is adequately characterized as not requiring prior NRC approval.

6.5.2 Containment Spray Systems

SRP Section 6.5.2 does not apply to the ABWR design.

6.5.3 Fission Product Control Systems (Includes Information Related to RG 1.206, Chapter 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System")

6.5.3.1 *Introduction*

This FSAR section addresses ESF systems designed to limit the release of radioactive materials from the containment into the environment following a LOCA or any other accident that releases fission products into the environment. The fission product control systems consist of the primary containment and the secondary containment. This section of the FSAR also addresses the pressure suppression pool as a fission product cleanup system.

6.5.3.2 *Summary of Application*

Section 6.5.3 of the STP Units 3 and 4 COL FSAR incorporates by reference Section 6.5.3 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52 ,Appendix A. In addition, in FSAR Section 6.5.3, the applicant provides the following:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

The departure eliminates the requirement to maintain equipment needed to mitigate a design-basis LOCA hydrogen release. The applicant has added the following:

> The primary containment atmosphere is inerted with nitrogen by the Atmospheric Containment System ACS [Atmosphere Control System]. The ACS is described in Subsection 6.2.5.

6.5.3.3 *Regulatory Basis*

The regulatory basis for the information incorporated by reference is in NUREG–1503. In addition, the relevant requirements of the Commission regulations for the fission product control systems, and the associated acceptance criteria, are in Sections 6.5.3 and 6.5.5 of NUREG-0800.

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

6.5.3.4 *Technical Evaluation*

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.5.3 of the certified ABWR DCD. The staff reviewed Section 6.5.3 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

The staff reviewed the information in the COL FSAR:

Tier 1 Departure

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

The technical evaluation of this departure is in Section 6.2.5 of this SER.

6.5.3.5 *Post Combined License Activities*

There are no post COL activities related to this section.

6.5.3.6 *Conclusion*

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

On the basis of the review of the STP COL application and the referenced DCD, including the Tier 1 information, the staff found that Section 6.5.3 of the ABWR DCD, including the information pertaining to the pressure suppression pool as a fission product cleanup system, is incorporated by reference in the STP COL application. The evaluation of STD DEP T1 2.14-1 is in Section 6.5.2 of this SER. The staff's review confirmed that the applicant has provided sufficient information to satisfy Sections 6.5.3 and 6.5.5 of NUREG–0800.

6.6 Preservice and Inservice Inspections and Testing of Class 2 and 3 Components and Piping (Related to RG 1.206, Section C.I.6.6, "Inservice Inspection of Class 2 and 3 Components")

6.6.1 Introduction

Inservice Inspection (ISI) Programs are based on the requirements of 10 CFR 50.55a, "Codes and Standards." Code Class 2 and 3 components, as defined in Section III of the ASME Boiler and Pressure Vessel Code (ASME Code), meet the applicable inspection requirements set forth in Section XI of the ASME Code, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components." ISI includes preservice (PSI) examinations before initial plant startup, as required by IWC-2200 and IWD-2200 of Section XI of the ASME Code.

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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

6.6.2 Summary of Application

Section 6.6 of STP Units 3 and 4 COL FSAR Revision 3 incorporates by reference Section 6.6 of the certified ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. In addition, in FSAR Section 6.6, the applicant provides the following:

Tier 1 Departures

- STD DEP T1 2.4-1 RHR System and Spent Fuel Pool Cooling

This departure adds the capability for a third RHR loop, RHR Division A, in the augmented fuel pool cooling and fuel pool makeup modes.

- STD DEP T1 2.4-3 RCIC Turbine/Pump

This departure replaces the RCIC turbine and pump system design with an integrated (monoblock) alternate turbine/pump system design.

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

This departure eliminates the requirement to maintain equipment needed to mitigate a design-basis LOCA hydrogen release.

Tier 2 Departure Not Requiring Prior NRC Approval

- STD DEP 6.6-1 Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

Section 6 of the ABWR DCD includes Section 6.6, "Preservice and Inservice Inspection of Class 2 and 3 Components and Piping," which addresses the guidelines of SRP Section 6.6, "Inservice Inspection and Testing of Class 2 and 3 Components."

- STD DEP 6.6-2 Erosion-Corrosion Program

Section 6.6.7.2 of the ABWR DCD is revised to reference the EPRI flow-accelerated corrosion program guidelines and also to clarify that this program applies to both single-phase and two-phase flows as discussed in the EPRI document.

COL License Information Items

The applicant provides the following information to address COL license information items, as described in ABWR DCD Section 6.6.9, "COL License Information."

- COL License Information Item 6.10 PSI and ISI Program Plans

This COL license information item addresses the applicant's commitment (COM 6.6-1) to prepare comprehensive plant-specific PSI/ISI Program plans. These plans will be submitted to the NRC at least 12 months before commercial power operation for the respective unit. The plan will be based on the final as-built plant configuration to address, for example, specific welds; bolting; and pipe supports. There will be a separate plan for Unit 3 and for Unit 4.

- COL License Information Item 6.11 Access Requirement

This license information item incorporates plans for nondestructive examinations (NDE) during the design and construction phases in order to meet all access requirements of the regulations, per IWC 2500 and IWD 2500 (Section 6.6.2).

6.6.3 Regulatory Basis

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

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

The regulatory basis and acceptance criteria of the resolution to COL License Information Items 6.10 and 6.11 are in Section 6.6 of NUREG–0800.

6.6.4 Technical Evaluation

As documented in NUREG–1503, NRC staff reviewed and approved Section 6.6 of the certified ABWR DCD. The staff reviewed Section 6.6 of the STP Units 3 and 4 COL FSAR and checked the referenced ABWR DCD to ensure that the combination of the information in the COL FSAR and the information in the ABWR DCD appropriately represents the complete scope of information relating to this review topic.[1] The staff's review confirmed that the information in the application and the information incorporated by reference address the required information relating to this section.

The evaluation of the Operational Program aspects of the ASME Code Class 2 and 3 PSI/ISI Program is discussed with Class 1 components in Section 5.2.4 of this FSER. Accordingly, this evaluation focuses on the acceptability of the COL applicant's supplemental information and responses to ABWR DCD COL license information items, as they relate to the PSI/ISI Program of the ASME Code Class 2 and 3 components.

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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 — Residual Heat Removal System and Spent Fuel Pool Cooling
- STD DEP T1 2.4-3 — Reactor Core Isolation Cooling System
- STD DEP T1 2.14-1 — Flammability Control System

These departures propose changes to the RHR, RCIC, and fire protection systems. The plant-specific departures for components are the listed systems in ASME Section XI, "Examination Category, Items Examined, and Exam Method." The NRC staff's review found that the information in the departures complies with the requirements in ASME Section XI. The departures are therefore acceptable, as they relate to FSAR Section 6.6. STD DEP T1 2.4-1 is evaluated in Section 5.4.7. STD DEP T1 2.4-3 is evaluated in Section 5.4.6. STD DEP T1 2.14-1 is evaluated in Section 6.2.5. These departures are also evaluated in Chapter 14.

Tier 2 Departures Not Requiring Prior NRC Approval

- STD DEP 6.6-1 — Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

This departure deletes the ABWR DCD text and states that the RHR heat exchanger nozzle to shell welds will be 100 percent accessible for the PSI during fabrication. The departure also adds text that discusses the NRC staff's review of ISI Program relief requests. 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 was unable to determine whether it is reasonable for this departure not to require prior NRC approval. The staff's review of this departure focused on the Operational Program aspects of the PSI and ISI. The staff was concerned that the departure provides a basis for not eliminating interferences during plant construction due to design, geometry, or materials of construction in order to enable the performance of ISI examinations, as required under 10 CFR 50.55a(g)(3)(ii). In order to determine whether it is reasonable for this departure not to require prior NRC approval, the staff issued **RAI 06.06-3** requesting the applicant to provide additional information.

In the response to this RAI dated July 23, 2009 (ML092080085), the applicant states that the STP Units 3 and 4 RHR heat exchanger will be designed to provide 100 percent accessibility for nozzle to shell welds for PSI, which should fully comply with 10 CFR 50.55a(g)(3)(ii). The deletion of the statement in STD DEP 6.6-1, as noted in the RAI, was intended to address certain changes in the ASME Code requirements relating to RHR nozzle to shell welds inspectability. The deletion was in no way intended to remove the requirement for 100 percent inspectability. On the basis of the applicant's response, the staff concluded that the regulation involving inspectability will be met, and RAI 06.06-3 is therefore closed. With respect to the impact of this departure on this section, the staff found it reasonable that the departure does not require prior NRC approval. The applicant's process for evaluating departures and other changes to the DCD is subject to NRC inspections.

- STD DEP 6.6-2 Erosion-Corrosion Program

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

COL License Information Items

- COL License Information Item 6.10 PSI and ISI Program Plans

NRC staff reviewed this COL license information item, which states (in part) that the initial ISI examinations conducted during the first 120 months of operation will comply, to the extent practical, with the requirements in ASME Code Section XI Edition and Addenda, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license, subject to the modifications listed in the regulations. COL License Information Item 5.2 states that the 1989 Edition of the ASME Code will be used. However, Table 1.8-21a states that the PSI/ISI Program will meet the requirements of the 2004 Edition of ASME Code Section XI. The specific version of ASME Code Section XI that is approved as the baseline code for the RPV design only is the 1989 Edition, as stated in the staff's FSER for the ABWR DCD. Later ASME Code Editions and Addenda are endorsed in 10 CFR 50.55a as the Code of Record for the ISI Program on a periodic basis. Lessons learned from plant operations and significant safety issues are incorporated into these later ASME Code Editions through a consensus approach with the industry. Because the staff's FSER of the related DCD approves the 1989 Edition of the ASME Code for the RPV design only, the staff was concerned that lessons learned were not incorporated during the COL applicant's development of the PSI/ISI Program. The staff therefore requested additional information in **RAI 06.06-2**.

In the response to this RAI dated July 23, 2009 (Ml092080085), the applicant notes that Subsection 5.2.6.2, "Plant Specific PSI/ISI," will be modified to state that the ISI/PSI Program will be based on the 2004 ASME Code Section XI, with no addenda (as identified in Table 1.8-21a). This code will be used to (1) select components for examinations, (2) identify components subject to examination, (3) describe the components exempted from examination by the applicable code, and (4) select isometric drawings used in the examinations. The staff concluded that revisions using the 2004 ASME Code for the PSI/ISI Program will incorporate lessons learned, which will result in a more robust design. The staff therefore considered the applicant's response acceptable. This RAI was tracked as **Confirmatory Item 06.06-2** in the SER with open items. The staff verified that Revision 4 of FSAR Subsection 5.2.6.2 reflects the changes discussed in the response to RAI 06.06-02. Confirmatory Item 06.06-2 is therefore closed.

COL License Information Item 6.10 states (in part) that the initial ISI examinations conducted during the first 120 months of operation should comply, <u>to the extent practical</u>, with the requirements in ASME Code Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b), on the date 12 months before the date of issuance of the operating license, subject to the modifications listed in the regulations. The regulation 10 CFR 50.55a(g)(3)(ii) requires Class 2 and 3 components and supports to be designed to enable the performance of ISI examinations during the initial 10-year interval. The text in 10 CFR 50.55a(g)(4) stating "<u>to the extent practical,</u>" only applies to ISI intervals subsequent to the first 120-month interval. The

COL applicant's use of the terminology "to the extent practical" implies that interferences with the performance of PSI and ISI examinations from design, geometry, and materials of construction will be addressed as impractical and the COL applicant may request relief from the examinations, which a licensee may do after the first 120 months of operation. The staff's expectation is that the regulations will be met during construction with respect to design to enable the performance of ISI examinations, and the components and coverage for PSI and ISI examinations required by the ASME Code will be accomplished with no requests for relief due to impracticality. With these concerns in mind, the staff requested additional information in **RAI 06.06-1**.

In the response to this RAI dated July 23, 2009 (Ml092080085), the applicant states that STP Units 3 and 4 will be fully compliant with the requirements of 10 CFR 50.55a, with regard to the PSI and ISI examinations. The applicant also states that the reference to the terminology "to the extent practical" appears in 10 CFR 50.55a(g)(4) and applies to 10 CFR 50.55a(g)(4)(i), which deals with ISI requirements for the initial 120-month ISI interval as well as the requirements in 10 CFR 50.55a(g)(4(ii), which address successive 120-month inspection intervals. The terminology "to the extent practical" applies to the initial and successive ISI intervals in effect and is endorsed after the ASME Code edition and addenda that are applied to the component design. On the basis of this information from the applicant, the staff concluded that the component design will meet the requirements of the ASME Code for the component design (2004), because the applicant is not requesting relief from the ASME requirements of the design Code of Record. However, the applicant also states that although no relief requests are expected to be submitted, it would be impractical to commit to no relief requests based on the ASME Code issued subsequent to the Code that is applied to the component design. The staff recognizes that there may be instances before fuel loading where relief may be necessary from ASME requirements. Therefore, the response is acceptable and **RAI 06.06-01** is resolved.

COL License Information Item 6.10 states (in part) that STPNOC will prepare a comprehensive plant-specific PSI/ISI Program plan, which will be submitted at least 12 months before commercial power operation for the respective unit and will be based on the as-built plant configuration, with separate plans for Unit 3 and Unit 4. The regulation in 10 CFR 50.55a(g)(4)(i) states that ISI examinations and pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code (or Code Cases) incorporated by reference in paragraph (b) of this section, on the date 12 months before the date scheduled for initial fuel loading under Part 52 of this chapter and subject to the limitations and modifications listed in paragraph (b) of this section. The supplemental information in COL License Information Item 6.10 meets the requirements of 10 CFR 50.55a(g)(4)(i) and is therefore acceptable.

The Flow Accelerated Corrosion (FAC) Operational Program (referred to as "erosion-corrosion" in the DCD) is an augmented inspection program included in the PSI/ISI Operational Program under FSAR Subsection 6.6.9.1. STP Units 3 and 4 COL FSAR Revision 2 did not provide sufficient detail for the NRC staff to determine whether lessons learned in NRC GL 89-08 are addressed in the Operational Program, such as the use of grid locations for the PSI and the basis for selecting components under the FAC Program. The staff therefore requested additional information in **RAI 06.06-4** about the program implementation schedule and consistency with the guidelines of the EPRI Nuclear Safety Analysis Center (NSAC) -202L, Revision 3, "Recommendations for an Effective Flow-Accelerated Corrosion Program." RAI 06.06-4 also asked for confirmation that the program will include preservice thickness measurements of susceptible components using grid locations and measurement methods most likely to be used for ISIs.

In the response to this RAI dated December 13, 2010 (ML103500239), the applicant clarifies that the FAC Program applies to both single-phase and two-phase conditions, which is consistent with the intent of GL 89-08. The response also states that the FAC Program for STP Units 3 and 4 will be implemented as part of the PSI/ISI Operational Program that will be submitted to the NRC at least 12 months before commercial power operation (per COL FSAR Table 13.4S-1). Furthermore, the response states that the program will follow the guidelines of EPRI-NSAC-202L Revision 3, including the selection of grid locations and consistency between PSI and ISI measurement methods and locations.

In the response to RAI 06.06-4, the applicant also proposes Standard DCD Tier 2 Departure STD DEP 6.6-2, which changes Subsection 6.6.7.2, "Erosion-Corrosion," as shown below:

- STD DEP 6.6-2

> Piping systems determined to be susceptible to ~~single-phase~~ erosion-corrosion shall be subject to a program of nondestructive examinations to verify the system structural integrity. The examination schedule and examination methods shall be determined in accordance with ~~the NUMARC program (or another equally effective program), as discussed in~~ Generic Letter 89-08, the guidelines of EPRI NSAC-202L Rev. 3, and applicable rules of Section XI of the ASME Boiler and Pressure Vessel Code.

The applicant's response includes a proposed description and evaluation of STD DEP 6.6-2 in the STP Units 3 and 4 Departures Report in Part 7 of the COL application. The applicant evaluates the departure in accordance with the requirements of 10 CFR Part 52, Appendix A, Section VII.B.5 and concludes that it does not require NRC approval. The applicant based this conclusion on finding no impact on ABWR DCD Tier 1, Tier 2*, TS, or the TS Bases section and on finding no detriment to safety because the departure adopts the latest industry FAC guidance.

The staff found the response acceptable because the applicant provided the information requested and committed to follow the EPRI/NSAC-202L guidelines and implement it in accordance with the PSI/ISI Program described in FSAR Tier 2, Table 13.4S-1. The EPRI guidelines are the current industry standard for addressing the FAC concerns expressed in GL 89-08. RAI 06.06-4 is resolved and is being tracked as **Confirmatory Item 06.06-4.**

- COL License Information Item 6.11 Access Requirement

COL License Information Item 6.11 states that "the plans for the NDE during the design and construction phases are incorporated in order to meet all access requirements of the regulations, per IWC-2500 and IWD-2500." As an integral part of the design process, the access requirements are incorporated into the applicable specifications. At the COL application stage, the PSI/ISI Program was not yet developed. This program will be developed during the construction phase. Although Section 6.6 applies to Class 2 and 3 components, these components still comprise portions of the PSI/ISI Program that includes Class 1 components. Therefore, RG 1.206 Section C.III.1, Chapter 5, C.I.5.2.4.1 for the reactor coolant pressure boundary (RCPB) applies. RG 1.206 states that the detailed procedures for performing the examinations may not be available at the time of the COL application, and the COL applicant should make a commitment to provide sufficient information demonstrating that the procedures meet ASME Code standards. This information should be provided at a predetermined time agreed upon by the applicant and NRC staff. The applicant identifies commitment COM 6.6-1 to provide the NDE procedures to the NRC staff. In order for the staff to obtain a reasonable assurance of the acceptability of the PSI/ISI Program, the staff must be able to inspect the plant during construction for conformance to the regulations and to the ASME Code of Record.

6.6.5 Post Combined License Activities

The applicant identifies the following commitment (COM):

- Commitment (COM 6.6-1) – The licensee will have the PSI and ISI Program plans available for NRC staff to review, including NDE procedures, to verify compliance with the ASME Codes and with other industry standards.

NRC staff will inspect the STP Units 3 and 4 PSI/ISI Program plans during construction to verify compliance with 10 CFR 50.55a requirements before fuel loading.

6.6.6 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 relating to the PSIs and ISIs, and testing of class 2 and 3 components and piping. With the exception of **Confirmatory Item 06.06-4**, 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 PSIs and ISIs and the testing of Class 2 and 3 components and piping that were incorporated by reference have been resolved.

In addition, the staff concluded that the COL applicant's proposed resolutions to the COL license information items under Section 6.6.9 of the STP Units 3 and 4 COL FSAR conforms to the relevant guidelines in SRP Section 6.6 and RG 1.206, Section C.III.1 and Chapter 6, C.I.6.6. The proposed resolutions are therefore acceptable. Conformance to these guidelines provides an acceptable basis for meeting the applicable requirements of 10 CFR 50.55a. On the same basis, the staff concluded that the COL applicant has adequately addressed COL License Information Items 6.10 and 6.11.

The staff found it reasonable that Departure STD DEP 9.4.2 is adequately characterized as not requiring prior NRC approval, per 10 CFR Part 52, Appendix A, Section VIII.B.5.

The staff found that the plant-specific information in STP Units 3 and 4 COL FSAR Revision 2, under Departures STD DEP T1 2.4-1, T1 2.4-3, and T1 2.14-1, conforms to the relevant guidelines in SRP Section 6.6, RG 1.206, Section C.III.1, and Chapter 6 C.I.6.6. The staff therefore found the information acceptable. Conformance to the RG 1.206 guidance provides an acceptable basis for meeting the applicable requirements of 10 CFR 50.55a.

As a result of the above confirmatory item, the staff was unable to finalize the conclusions relating to the PSIs and ISIs and the testing of Class 2 and 3 components and piping, in accordance with the NRC requirements.

6.7 High Pressure Nitrogen Gas Supply System

Section 6.7 of the STP Units 3 and 4 COL FSAR incorporates by reference, with no departures or supplements, Section 6.7, "High Pressure Nitrogen Gas Supply System," of the ABWR DCD Revision 4, referenced in 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no outstanding issue related to this 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 high pressure nitrogen gas supply system have been resolved.

6.8 References

The following are references used in Section 6.2

6.2-1 W. J. Bilanin, "The General Electric Mark III Pressure Suppression Containment System Analytical Model," Licensing Topical Report, NEDO-20533, June 1974.

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6.2-3 General Electric Energy Nuclear, "ABWR Containment Analysis," Licensing Topical Report, NEDO-33372, September 2007.

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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.

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