PMSTPCOL NPEmails

From: Joseph, Stacy
Sent: Friday, May 14, 2010 11:25 AM
To: 'Tomkins, James'; 'ccchappell@stpegs.com'
Cc: STPCOL
Subject: PROPRIETARY - Chapter 6 SER with OI
Attachments: ML1013105150.pdf; ML1013105198.pdf

Jim and Coley,

Attached is the Revised Chapter 6 SER with Open Items with the Proprietary Info removed. Please perform a Proprietary Review to ensure all proprietary information has in fact been removed.

Thank you,
Stacy

Hearing Identifier: SouthTexas34NonPublic_EX
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Subject: PROPRIETARY - Chapter 6 SER with OI
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From: Joseph, Stacy

Created By: Stacy.Joseph@nrc.gov

Recipients:
"STPCOL" <STP.COL@nrc.gov>
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6.0 ENGINEERED SAFETY FEATURES

This chapter of the Final Safety Analysis Report (FSAR) discusses the design and functional requirements of engineered safety features (ESF) 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 operation of the ESF. Materials have been selected to withstand the environmental conditions encountered during normal operation 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 request for additional information (RAI) 06.01.01-1, the applicant deletes 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).

[redacted]

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, U.S Nuclear Regulatory Commission (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 responded in a letter dated January 28, 2010. The applicant's response proposes a revision to 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

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.

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 meeting SA-351 Grades CF3M, 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. The staff found these material specifications acceptable. These material specifications and grades are 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 will verify that the applicant has modified FSAR Subsection 6.1.1.1.1 and Table 6.1-1, as discussed above, in STP FSAR Revision 4. This issue is being tracked as **Confirmatory Item 06.01.01-1**.

6.1.1.5 *Post Combined License Activities*

The applicant identifies the following commitment:

- COM 6.1-1 – This commitment requires the applicant to provide, in the FSAR, materials that will be used in the Reactor Building Cooling Water System heat exchanger and the Reactor Service Water System pumps and valves prior to the initiation of the respective unit preoperational testing.

As discussed above, the applicant intends to delete COM 6.1-1 because it will include this information in the FSAR Revision 4.

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. With the exception of **Confirmatory Item 06.01.01-1**, no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the 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. However, as a result of the **Confirmatory Item 06.01.01-1**, the staff was unable to finalize the conclusions relating to the metallic materials in accordance with the NRC requirements.

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. 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

[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.

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:

COL License Information Item

- 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.
>
> (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, the applicant summarized how the evaluation of combustible gas generation for the non-conforming coatings and organic materials in the containment will be performed in accordance with 10 CFR Part 50, Appendix B. The summary identified 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 also proposed 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 is being tracked as **Confirmatory Item 6.01.02-1.**

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. With the exception of **Confirmatory Item 6.01.02-1**, no outstanding information is expected to be addressed in the COL FSAR related to this section. Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A ,Section VI.B.1, all nuclear safety issues relating to the 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. However, as a result of **Confirmatory Item 6.01.02-1**, the staff was unable to finalize the conclusions relating to the organic materials in accordance with the NRC requirements.

6.2 <u>Containment Systems</u>

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 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 in 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.

[redacted]

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 m x 2 m (3.3 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 (kPa) (45 pounds per square inch gauge [psig]) and 171 °C (340 °F), respectively. The design drywell-to-wetwell differential pressures are +172.4 kPa (25 psig) and -13.8 kPa (-2 psig). The design drywell-to-reactor building negative differential pressure is -13.8 kPa (-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 center lines 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 low water level.

The wetwell is designed for an internal pressure of 310 kPa (45 psig) and a temperature of 103.9 °C (219 °F). The design wetwell-to-reactor building negative differential pressure is -13.8 kPa (-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.

[redacted]

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 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 consistent with normal plant operation 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 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:

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

- 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 then to trip the condensate pumps.

- STD DEP T1 2.4-3 — Reactor Core Isolation Cooling (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 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, thus necessitating the use of 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 of 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 Subsection 6.2.1.1.7, Section 6.2.1, Subsection 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” indicating a reactor vendor change.

<u>COL License Information Items</u>

- 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 (SRP).

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 is based on compliance with 10 CFR 50.46(b)(5), as it related 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 is 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 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, please refer to COL application Part 07, Section 5.0 for a listing of all FSAR sections affected by these Tier 1 departures. In addition, compliance with 10 CFR Part 52, Appendix A, Section VIII.A.4 for Tier 1 departures will be addressed by the staff in a future exemption evaluation.

- 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 containment. Additionally, the departure also implements a condensation pump 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. Staff considers the implementation of this departure to 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, and therefore is acceptable.

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

- 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 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.

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

The 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 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. For the purposes 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 2.14-1 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.

The design change incorporates changes to regulations that occurred after the issuance of the design certification for the ABWR. This departure 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 finds 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 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 topical 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 release 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. NRC staff raised the above issue in **RAI 06.02.01.01.C-14**, which is being tracked as **Open Item 06.02.01.01.C-14**.

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 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 was performed using the DCV loss coefficients and considering 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. This total effective vent loss was applied 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 the FSAR accordingly. This RAI is being tracked as **Open Item 06.02.01.01.C-15**.

The applicant also states that the decay heat curves used in the DCD analysis are based on the best estimate of the American National Standards Institute (ANSI)/American Nuclear Society (ANS)-5.1 (1979) and are considered to be non-conservative for the long-term containment loading analysis. To address this non-conservatism, the decay heat curves used in the updated containment analysis were revised to reflect the 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. Reference 6.2-2 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 non-conservative 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. Reference 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 (210.3 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. This RAI is being tracked as **Open Item 06.02.01.01.C-18**.

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 has increased the design suppression pool temperature limit to 100 °C, and has accordingly revised the ECCS pump NPSH calculations by assuming the updated suppression pool design temperature limit. The staff finds such an approach acceptable.

The GOTHIC-calculated drywell-to-wetwell peak differential pressure is 146 kPaG, which provides a design margin of 18 percent, when compared to the design drywell-to-wetwell differential pressure is 172.6 kPaG. The staff issued **RAI 06.02.01.01.C-16** requesting the applicant to update the FSAR accordingly. This issue is being tracked as **Open Item 06.02.01.01.C-16**.

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 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 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 provided the requested information in a response dated January 20, 2010 and stated 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. The Staff updated the MECLOR model based on the detailed main steam line design data provided in the January 20, 2010 response 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 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. **RAI 06.02.01.01.C-10** requested 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 MELCOR confirmatory base-case calculation for the long-term MSLB. This resolved **RAI 06.02.01.01.C-10**.

- The STP Units 3 and 4 containment analysis presented in Reference 6.2-2 does not document the results of the long-term FWLB accident simulation. Therefore the staff issued **RAI 06.02.01.01.C-11** requesting the results of containment pressure/temperature analysis and M&E release rate to containment for the long-term FWLB accident simulation. The applicant provided the requested information in a response dated January 21, 2010. This resolved **RAI 06.02.01.01.C-11**.

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 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 (prior to 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 pools swell analysis based on the relevant test data rather than on the approval of the PICSM 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 new alternate method by comparing it against one MARK III PSTF test (test 1 from the PSTF series 5806). The NRC 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 the topical 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 specified as a boundary condition to the GOTHIC model. The applicant documents the results of the comparison of the GOTHIC model with the ABWR DCD pool swell analysis and presents the revised ABWR pool swell analysis in Reference 6.2-4. For the comparison against the DCD methodology, the GOTHIC model used the same drywell pressure boundary condition 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/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 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 the PSTF series 5806 and test 10 from the 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. This resolved **RAI 06.02.01.01.C-12**.

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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. The applicant’s analysis showed that the pool swell and swell velocity are not very sensitive to node size. This resolved **RAI 06.02.01.01.C-13**.

Tier 2 Departures Not Requiring Prior NRC Approval

- 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 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, Section 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/foot instead of seconds/meter, as required in this case. This departure corrects the multiplying factor from 0.0016 seconds/foot to 0.0052 seconds/meter.

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. Also, the applicant states that the correct loads are used for the structural analyses to show that the structures and components 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 net positive suction head 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 are subject to NRC inspections.

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

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, 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 of 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, 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 residual heat removal (RHR) pumps and one HPCF pump would be in operation and each pump has two strainers, the total blocked area of 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: GL-04-02 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. In response, in a letter dated April 14, 2010, the applicant agreed to update STP 3&4 FSAR Section 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 addresses the staff's concern and therefore is acceptable. This is being tracked as **Confirmatory Item 06.02.02-26.**

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 containment. 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 containment. 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 because while the applicant has committed to eliminate all fiber insulation from containment, it has assumed one cubic foot of fiber in its analysis and is thus conservative.

During the audit (ML092370709), the applicant stated that the thermal insulation in the STP Units 3 and 4 primary containment 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 3D 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. URG guidance 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 propose a plan to 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 analysis and found this assumption acceptable.

[redacted]

According to the response to the 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 TEPCO on quantities of material obtained from suppression pool cleanup systems at Kashiwazaki-Kariwa Units 6 and 7, which are the oldest operating ABWRs. The staff reviewed these data and determined that the quantities of rust flakes and sludge that were assumed for the STP Units 3 and 4 ECCS suction debris strainer design are greater than quantities from TEPCO and thus are 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 is conservative.

The applicant's supplemental response to **RAI 06.02.02-6** dated October 29, 2009, 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, 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 as the design 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 non-conservatism. The non-conservatism 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 effect of conservative and non-conservative 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 net positive suction head (NPSH). The staff requested the applicant to update the FSAR to include the non–conservatism. **This is being tracked as Confirmatory Item 06.02.02-20.**

It was not clear to the staff from applicant's submittals how 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 alternative approach. In a supplemental response dated October 29, 2009, 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, 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

[redacted]

RAI 06.02.02-6 are discussed below.

Following the audit (ML092370709) 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 letters responding to **RAI 06.02.02-6** dated October 29, 2009, and February 15, 2010, the applicant submits References 6.2-5 through 6.2-7 and summarizes relative conservatisms and non-conservatisms in the STP Units 3 and 4 design compared to RJABWR. Relative conservatisms identified by the applicant include the following:

- RJABWR contains fiber insulation on small bore piping in the zone of influence, which is transported to the suppression pool. STP 3&4 design does not allow fiber insulation in the containment and fiber on small bore piping will be replaced with reflective metallic insulation (RMI). Head loss on the debris strainer due to RMI without fiber is negligible. The only fiber postulated to be in the STP 3&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.
- RJABWR contains calcium silicate insulation which is transported to the suppression pool. Only RMI is allowed for thermal insulation inside STP 3&4 primary containment. Calcium silicate is a significant contributor to head loss and STP 3&4 will eliminate this contributor to head loss.
- RJABWR ECCS suction debris strainer design did not account for miscellaneous latent debris (equipment tags, tape, and stickers or placards affixed by adhesives) because URG guidance that RJABWR used does not have a requirement to account for such debris. STP 3&4 ECCS suction debris strainer design accounts for miscellaneous latent debris.

A relative non-conservatism is that the RJABWR ECCS suction debris strainer design assumes a pump design flow rate, but the STP Units 3 and 4 design will assume a pump runout flow rate resulting in a higher head loss. In Reference 6.2-7, the applicant evaluated the impact of this non-conservative change and the above conservative changes on the STP Units 3 and 4 ECCS debris strainer design. This evaluation assumed only a partial replacement of fiber insulation in the STP Units 3 and 4 containments with RMI, which assumed that 0.73 m^3 (25.8 ft^3) of fiber debris would be transported to the strainers. After performing the calculation in Reference 6.2-7, 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 Reference 6.2-7 shows that NPSH margins exist for both the RHR and HPCF pumps. After reviewing Reference 6.2-7, the staff determined that based on the conservative and non-conservative changes cited above, NPSH margins exist for the RHR and HPCF pumps.

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

Section 6C.2 of the STP Units 3 and 4 FSAR states that "[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 continuous addition of water into the containment. Therefore, the staff issued **RAI 06.02.02-3** requesting the applicant to explain how it accounted for such pressurization. In response, in a letter dated September 28, 2009, the applicant's response states that ACIWA mode of RHR for reactor vessel injection and drywell spray is analyzed in Appendix 19E.2.2 of the ABWR DCD. The operator actions associated with 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) presented in FSAR Appendix 18A. These operator actions included in the EPGs include precautions to maintain the primary containment water level and pressure low enough to preclude primary containment failure and to terminate injection when required. The staff found that the applicant's response addresses the staff's concern and is acceptable because the operator's actions are included in the EPGs which conform to regulatory position 2.2 of R.G 1.82, Revision 3. Therefore, **RAI 06.02.02-3** is closed.

ABWR DCD Tier 1 refers to a 50 percent blockage of pump suction strainers in determining the NPSH margin, as stated in RG 1.82 Revision 1. However, STP Units 3 and 4 are committed 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, 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, and **RAI 06.02.02-22** is closed.

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

Protective coatings (i.e., paints) are a potential source of LOCA-generated debris in containment. Such debris could potentially contribute to plugging of ECCS suction strainers, downstream components, and fuel. The amount and size of the debris depends on the type, location, and condition of the coating. The potential for such debris to degrade emergency core cooling has been 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 committed to following the guidance related to ECCS blockage in RG 1.82 and 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 was documented in an SER dated September 3, 1998 (Ref. 6.2-9). In addition, the staff used coatings evaluation guidance issued in March 2008 for resolution of

Generic Letter 2004-02 regarding potential debris blockage of PWR emergency recirculation (Ref. 6.2-10). This document supplements the RG 1.82 and URG on several topics, including coating debris particle size. Conforming to this guidance provides one acceptable way to meet the requirements of 10 CFR 50.46 and address the concerns expressed in GL 2004-02 related to coatings debris.

In a 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, the applicant states that STP assumes a 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 did not state whether the URG was conservative for STP Units 3 and 4 or describe a size distribution for the coatings. The need to determine coating debris size for BWRs is listed in Regulatory Position 2.3.1.4 of RG 1.82. For evaluating suction strainers, the URG assumes 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 coating debris and the particle size distribution.

In a response, dated September 22, 2009, the applicant clarifies the basis for assuming 85 pounds coating 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 85 pounds applies to epoxy/inorganic coating systems while 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, the type of coating systems specified in the ABWR DCD. Therefore, the staff found this coating debris quantity acceptable because the applicant's analysis conforms to the URG guidance for coatings quantity.

With respect to the particle size, the head loss testing for the ECCS screens assumed 85 pounds of coating 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 the February 22, 2010 response to **RAI 04.04-3** regarding downstream effects testing for fuel assemblies. In the response, the applicant states that the 85 pounds coating debris is assumed to be entirely fine particles. This approach assumes that all coating 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 (Ref. 6.2-10), states that where there is a possibility of forming a thin fiber bed, coatings debris should be treated as fine particles (which can be trapped by the bed and contribute to head loss). Since 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 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.3.1.8.3, which states the coatings debris for downstream fuel effects testing will be small particles and they are assumed to pass through the ECCS strainers. This testing is required by **License Condition 04.04.03.** The proposed FSAR revision includes a table that identifies the coating debris quantity for the testing, which the applicant determined from scaling

[redacted]

the coating debris quantity for the plant (85 pounds) down to a single fuel assembly and then adding a 10-percent margin. The staff confirmed that the coating debris quantity identified for the testing was calculated according to the applicant's description. The staff found the particle size distribution acceptable because the applicant's approach of assuming the coating debris distribution to be entirely fine particles conforms to the staff's guidance in Reference 6.2-9 that supplements RG 1.82 and the URG, as described above. The proposed changes to the FSAR are being tracked as **Confirmatory Item 04.04-3.**

Therefore, for the reasons described above, the staff determined the applicant's treatment of coatings debris acceptable, including the coatings debris quantity and assumed particle size because it conforms 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. 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. The NRC staff published detailed guidance in March 2008 for PWR licensees to evaluate plant-specific chemical effects (Ref. 6.2-10). This includes guidance on using WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191" (Ref. 6.2-11). Separately, the staff issued an SER to approve, with limitations and conditions, the use of WCAP-16530-NP 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.

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 representative of a BWR post-LOCA environment has not been thoroughly studied. Because chemical debris generation may depend on 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 Japanese reference BWR (Reference 6.2-11). 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 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:

[redacted]

- 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 in letters dated September 28, 2009 (ML092730448), December 22 2009 (ML093580193), January 28, 2010 (ML100330402), and February 22, 2010 (ML100560113). These responses state that there will be no calcium, silicon, or phosphate in the insulation in 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 would be inconsequential due to the absence of phosphate and silicon. However, since concrete dissolution can generate elements affecting chemical precipitates (e.g., sodium aluminum silicate), the staff requested that the applicant address the contribution of exposed concrete to chemical debris formation as part of **RAI 06.02.02-27**. Therefore, the significance of calcium and silicon dissolved from exposed concrete within the coatings ZOI is being tracked as part of **Open Item 6.02.02-27.**

The applicant's February 22, 2010 response describes calculations performed to support the assumption of no chemical precipitation. The response also includes a proposed revision to FSAR Section 6C.3. The FSAR revision states that because aluminum is prohibited in purchase specifications, no aluminum is expected in 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 to evaluate aluminum corrosion and precipitation. These calculations assumed corrosion of the aluminum according to the release rate equations in WCAP-16530-NP. The calculations also compared the total amount of dissolved aluminum to the solubility limit to determine whether it would remain dissolved or precipitate as solids at the applicable pH and temperature. This part of the evaluation was based on the solubility data in the report, "Aluminum Solubility in Boron Containing Solutions as a Function of pH and Temperature" (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 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, corresponding to the LOCA conditions described in DCD Section 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 it was available to dissolve throughout the 30-day period. The value of 4.5 square feet proposed in the FSAR corresponds to the pH 5.3 condition, which had the lowest solubility limit. The evaluations for higher pH values and for pH 5.3 with a higher final suppression pool temperature (45°C), produced aluminum surface areas as high as 3000 square feet. Therefore, the proposed FSAR revision states that the implementation of the Suppression Pool Cleanliness and Foreign Material Exclusion Programs will ensure that the quantity of latent aluminum will be less than 4.5 square feet.

The staff reviewed the applicant's analysis (Supplemental response No. 2 to **RAI 06.02.02-11**) and determined the response is not complete. The aluminum corrosion calculations and solubility data used to analyze chemical effects were based on boron-containing solutions. These analysis tools do not apply directly to boron-free BWR coolant. In addition, the analysis did not include all relevant chemical debris sources. Therefore, the staff issued **RAI 06.02.02-27** requesting the applicant to provide the following information:

- Analysis of aluminum chemical effects using corrosion and solubility data applicable to the post-LOCA ECCS fluid at STP 3&4.
- If the pH is expected to vary with time during the postulated 30-day post-LOCA period, provide an analysis of the chemical effects based on the predicted transient or explain how your approach is bounding. (For example, addition of sodium pentaborate from the standby liquid control system would increase pH over some time period.)
- Discuss your plans to address chemical effects not considered in the initial analysis, such as:
    - Constituents dissolved from concrete in the coatings zone of influence (ZOI), since the NRC coatings guidance assumes removal of the coating within the ZOI. Concrete dissolution generates elements that can form chemical precipitates, including precipitates containing aluminum (e.g., sodium aluminum silicate).
    - Zinc, which corroded at a low rate in testing related to PWRs but would be expected to corrode at higher rates in neutral and acidic solutions. This may result in levels of zinc particulate, zinc corrosion products, and zinc in solution that could contribute to other chemical precipitates.
    - Corrosion products (iron oxide) resulting from iron or steel corrosion prior to or following a LOCA
    - Any other material present in containment that would be exposed to the post-LOCA fluid and has not been addressed by an integrated chemical effects analysis for the ABWR environment.
- If your analysis predicts the formation of chemical debris, discuss your plans for addressing the impact of this debris on the ECCS strainers and fuel assemblies (e.g., integrated strainer testing or a simplified approach that relies on significant clean screen area).

Since the staff has not completed the review of the applicant's calculations and proposed FSAR revisions this item is being tracked as part of **Open Item 6.02.02-27.** The staff will determine if the applicant's evaluation is acceptable based on conformance to RG 1.82, Subsection 2.3.1.8, on chemical debris formation from the chemical and thermal environment in the containment pool.

The evaluation of ECCS components downstream of the suction strainers is meant to addresses blockage of flow paths, wear, and abrasion (e.g., valves, pumps components and heat exchanger tubes), and blockage of fuel assembly flow channels. RG 1.82, Revision 3, Subsection 2.1.2.2 addresses the need to prevent clogging of flow restrictions and damage from fine particles downstream of passive strainers. In an SER dated December 2007, 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 April 29, 2009, response to **RAI 06.02.02-1**, the applicant proposed 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," and that this evaluation would 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 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.

Since WCAP-16406-P was developed for PWRs, the staff, in **RAI 06.02.02-10**, asked the applicant to justify applying it to a BWR. In the September 28, 2009, response, the applicant states that the WCAP-16406-P methodology was applicable to STP Units 3 and 4 based on the similar materials and clearances for BWR and PWR downstream components. In the December 21, 2009, response to **RAI 06.02.02-13**, 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 that 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 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. Since 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 safety evaluation. 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 SER on WCAP-16406 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 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 three 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 3 percent flow area acceptable because it is within valve manufacturing and fluid-flow calculation tolerances.

All valves over 1.5 inches, and nearly all valves smaller than that, are evaluated for plugging. Some valves are in the closed position during the event 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 acceptable in the SER for WCAP-16406-P because it conforms to the staff's SER for NEI 04-07. To summarize for valves, since 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 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 it 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 SER. Therefore, in accordance with COM 6C-1, the applicant will need to review the staff's evaluation of the WCAP and determine if limitations and conditions apply. The proposed revisions, including FSAR Section 6C.3.2 and COM 6C-1 are being tracked as **Confirmatory Item 6.02.02-27.**

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 design margin and is acceptable.

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

- 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. NRC 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.

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

- 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. In the letter, dated September 28, 2009, the applicant's response to **RAI 06.02.02-7** 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, 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 suppression pool cleanup 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 the suppression pool cleanliness.

The applicant's response also states that NUREG-1434, "BWR Standard Technical Specifications, General Electric Plants BWR/6" does not include a surveillance on suppression pool cleanliness and 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 Criteria. 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 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 implementation. In response to this RAI, the applicant agreed to update the FSAR by adding Subsection 6.2.1.7.1 on 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 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 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 addresses staff's concern 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 providing 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 is being tracked as **Open Item 06.02.01.01.C-17.**

6.2.1.5 *Post Combined License Activities*

The applicant identifies the flowing commitment:

- Commitment (COM 6C-1) – Perform 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.

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. With the exception of the above open items, 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 in other NRC RGs. The applicant has adequately addressed COL License Information Item 6.4. The staff's conclusion License Information Item 6.5 is pending resolution of the open item discussed above.

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, and, therefore, complies with 10 CFR 50.46(b)(5) as it relates to debris protection for ECCS strainers.

However, as a result of the above open items, the staff was unable to finalize the conclusions relating to containment functional design, in accordance with the NRC requirements.

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 Revision 4 of the ABWR DCD, which is itself incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remains for review.[1] The staff's review confirmed that there is no

[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.

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, which is itself incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this 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 secondary containment functional design have been resolved.

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. In addition, the applicant provides the following:

<u>Tier 1 Departure</u>

- 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:

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 changed the implementation architecture
1. Testing and surveillance changes

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

- 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 the correction 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 has identified 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

[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.

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 collectively ensure that the design fully complies with NRC rules and regulations, and therefore does not impact the probability of the occurrence of accidents or the consequence of accidents, and does not create accidents of a different type that has 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 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.

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 and concludes 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*

FSAR Section 6.2.5, “Combustible Gas Control in Containment,” 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 in excess of 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:

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

- STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

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

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

- 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 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 requiring prior NRC approval 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:

Tier 1 Departure

- 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 flammability control system (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 atmospheric control system (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 for controlling combustible gases added after the issuance of the design certification for 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.

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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.

6.2.5.5 *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. The 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 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 Control 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 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

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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.

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 primary containment valves and components (Type C), in accordance with requirements of 10 CFR Part 50, App. 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, App. J.

Supplemental Information

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

Implementation schedules and milestones were provided by the applicant 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 that the schedule for the development of the procedures as well as their scope reasonable. The staff found the information in COL FSAR Section 13.4S for Operational Program Implementation of the Containment Leak Rate Testing Program acceptable based on the requirements of 10 CFR Part 50, Appendix J.

6.2.6.5 *Post Combined License Activities*

There are no post COL activities related to this section.

6.2.6.6 *Conclusion*

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 application to the relevant NRC regulations; acceptance criteria defined in NUREG–0800, Section 6.2.6 and 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 can be 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 Emergency Core Cooling Systems

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 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 fuel pool cooling (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 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 2.2m to 1.7m and 2.4m to 2.0m, 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 ECCS subsystem 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.

[redacted]

- 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 analysis results for 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.

The staff reviewed the information in the COL FSAR:

<u>Tier 1 Departures</u>

- 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

[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.

[redacted]

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 safety evaluation report (SER). An evaluation of the changes to the Inspections, Tests, Analyses, and Acceptance Criteria (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 the subsections 6.3.3.2 and 6.3.3.10. NRC staff found that these administrative departure does not affect the presentation of any design discussion or qualification of design margin and is acceptable.

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

In a letter dated May 13, 2009 (U7-C-STP-NRC-090044), the applicant’s response to **RAI 06.03-1** provides the technical bases for the change. The applicant states that the pressure head across the pump for the low-pressure flow of 727 m^3 is according to the following equation:

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

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/h at a differential pressure of 0.69 MP_a. DCD Tier 2 Table 6.3-8 specifies this flow at 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 that it is reasonable that this departure does not require prior NRC approval. The applicant’s response to this RAI acceptable, and **RAI 06.03-1** is closed.

FSAR Tables 6.3-8 and 6.3-9

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

COL License Information Item

- COL License Information Item 6.6 ECCS Performance Results

The current licensing basis for the STP Units 3 and 4 includes the fuel design as 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, is 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 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 results 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 Technical Specifications 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. The 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 application to the relevant NRC regulations and acceptance criteria outlined in NUREG–0800, Section 6.3 and in 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 the 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.

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.

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

COL License Information Item

- 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, acetic acid (water transport); gasoline (water transport); sodium hypochlorite (onsite storage); 1-hexene (offsite storage); and acetic acid (offsite storage) pose toxic gas threats to the control room. 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**, 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 due to 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 is being tracked as **Open Item 06.04-2**.

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

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. The 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. With the exception of **Open Item 06.04-2**, 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 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 does not address the toxic gas evaluation for control room habitability. This is being tracked as **Open Item 06.04-2**.

As a result of **Open Item 06.04-2,** the staff was unable to finalize the conclusions relating to the control building HVAC, in accordance with the NRC requirements.

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 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:

<u>COL License Information Items</u>

- 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 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 is 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 engineered safety features 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 identified 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 Information Items 6.9 and 6.9a are satisfied based on (1) meeting the requirements of 10CFR100, "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 10CFR 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, 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 06.05.0-1**, which requested 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 perform a physical drawdown test of the reactor building to demonstrate the SGTS capability. The applicant's response to **RAI 06.05.0-1** clarified that this analysis would not eliminate the physical drawdown testing required by Technical Specification Surveillance Requirement SR 3.6.4.1.4. The staff found the applicant's response to this RAI (dated July 27, 2009) addressing this COL information item acceptable and consistent with 10 CFR 52.79 and 10 CFR 50.36 with regard to technical specifications surveillance requirements. Therefore, **RAI 06.05-1** is closed.

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

NRC staff issued **RAI 06.05.01-2** (eRAI 3026), which requested 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** stated that laboratory testing of a representative carbon adsorber sample would be performed if the SGTS exceeds 90 hours of operation per year, in accordance with RG 1.52 Revision 2 and Technical Specification 5.5.2.7. The staff found the applicant's response to the RAI addressing this COL information item acceptable and consistent with regulatory guidance on assuring filtration system efficacy and filter testing. In accordance with the applicant's response to **RAI 06.05.01**, the applicant stated that it will revise FSAR Subsection 6.5.5.2 to clarify that the ability of the SGTS to perform its intended function will be demonstrated by appropriate laboratory testing. The resolution of this RAI is being tracked as **Confirmatory Item 6.5-1**.

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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.

[redacted]

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 Technical Specification Surveillance Requirement 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. With the exception of **Confirmatory Item 6.5-1**, no outstanding information is expected to be addressed in the COL FSAR related to this section Pursuant to 10 CFR 52.63(a)(5) and Part 52, Appendix A, Section VI.B.1, all nuclear safety issues relating to the engineered safety features filter systems that were incorporated by reference have been resolved.

The staff compared the information in the application to the relevant NRC regulations, 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 has 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.

However, as a result of **Confirmatory Item 6.5-1,** the staff was unable to finalize the conclusions relating to the fission products removal and control systems, in accordance with the NRC requirements.

6.5.2 Containment Spray Systems

SRP Section 6.5.2 does not apply to the ABWR design;

[redacted]

[redacted]

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, which is addressed in Section 6.5.5 of NUREG–0800.

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 ABWR DCD Revision 4, referenced in 10 CFR 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 identified 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

[redacted]

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 addressed the relevant information, and no outstanding information is expected to be addressed in the COL FSAR related to this section.

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 American Society of Mechanical Engineers (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)

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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

examinations before initial plant startup, as required by IWC-2200 and IWD-2200 of Section XI of the ASME Code.

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 in Revision 4 of the ABWR DCD. 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.”

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 license information item addresses the applicant’s commitment to prepare comprehensive plant-specific PSI and 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. (COM 6.6-1)

- 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 identified 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 and ISI Programs 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 PSI and ISI Programs 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. 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 to not require prior NRC approval, the staff issued **RAI 06.06-3** requesting the applicant to provide additional information.

In a letter dated July 23, 2009, the applicant's response to the RAI 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. The staff concluded from the applicant's response that the regulation involving inspectability will be met. The RAI is therefore considered 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.

<u>COL License Information Items</u>

- COL License Information Item 6.10 PSI and ISI Program Plans

NRC staff reviewed this departure, 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 and ISI Programs 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 ISI Programs 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 and ISI Programs. The staff therefore requested additional information in **RAI 06.06-2**.

In a letter dated July 23, 2009, the applicant's response to the RAI 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 Boiler and Pressure Vessel 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 will incorporate lessons learned, which will result in a more robust design. The staff therefore considered the applicant's response acceptable. This RAI is being tracked as **Confirmatory Item 06.06-2.**

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 "<u>to the extent practical</u>" infers 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 a letter dated July 23, 2009, the applicant's response to the above RAI 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 to 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 the **RAI 06.06-01** is resolved.

COL License Information Item 6.10 states (in part) that STPNOC will prepare a comprehensive plant-specific PSI and 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 and 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 Generic Letter 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**.

In a letter dated September 15, 2009, the applicant's response to the RAI states the following:

> The STP 3&4 FAC program will follow the guidelines of EPRI NSAC 202L, the program will be implemented as part of the PSI and ISI programs, separate plans for Unit 3 and Unit 4 will be submitted at least 12 months prior to commercial operation as part of the PSI and ISI submittals, baseline thickness measurements will be included in PSI using grid locations in the EPRI NSAC

202L guidelines, and the PSI will use techniques expected to be used in ISI (ultrasonic and possibly radiographic).

The response also includes Subsection 6.6.7.2, “Erosion-Corrosion,” which is new and will be included in Revision 3 of the COL application FSAR. This information was initially incorporated by reference from the ABWR DCD. The paragraph that will be added to the COL application is a future revision of FSAR Subsection 6.6.7.2. This proposed subsection includes the Electric Power Research Institute (EPRI)/ Nuclear Safety Analysis Center (NSAC) 202L guidelines as a basis for the FAC Program schedule and examination methods. The staff found this proposal acceptable because the EPRI guidelines are the current industry standard for addressing the FAC concerns expressed in GL 89-08. This RAI 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 and ISI Programs were not yet developed, but they will in fact 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 and ISI Programs that include 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. The RG 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. In order for NRC staff to obtain a reasonable assurance finding of the acceptability of the PSI and ISI Operational Programs, the staff must be able to inspect the plant during construction for conformance to the regulations and the ASME Code of Record.

6.6.5 Post Combined License Activities

The applicant identifies the following commitment (COM):

- 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 and 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. 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 the PSIs and ISIs, and testing of class 2 and 3 components and piping. With the exception of Confirmatory

Items 06.06-2 and 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 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 STP Units 3 and 4 COL FSAR conforms with 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 closed out 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 STD DEP T1 2.4-1, T1 2.4-3, and T1 2.14-1 conforms 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 guidelines provides an acceptable basis for meeting the applicable requirements of 10 CFR 50.55a.

6.7 <u>High Pressure Nitrogen Gas Supply System</u>

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 Revision 4 of the ABWR DCD, which is itself incorporated by reference into 10 CFR Part 52, Appendix A. NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this 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 <u>References</u>

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.

6.2-2 Westinghouse Electric Company, LLC, "Implementation of ABWR DCD Methodology Using GOTHIC for STP 3 and 4 Containment Design Analyses." WCAP-17058, June 2009.

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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.

[redacted]

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

6.2-4 Toshiba Corporation, "ABWR Pool Swell Calculation Methodology Using GOTHIC," UTLR-0005, September 2009.

6.2-5 The Evaluation Report for Net Positive Suction Head of Pump in Emergency core Cooling System, Proprietary, STP Doc. U7-RHR-M-RPT-DESN-0001, Rev. B, February 10, 2010.

6.2-6 The Supplementary Documentation for the Head Loss Evaluation Report of Japanese ABWR ECCS Suction Strainer, Proprietary, STP Doc. U7-RHR-M-RPT-DESN-0002, Rev. C, February 10, 2010.

6.2-7 The Evaluation Example of the Head Loss of the ECCS Suction Strainer and Pipe in the ECCS Pump Run-out Flow Condition, Proprietary, STP Doc. U7-RHR-M-RPT-DESN-0003, Rev. A, May 27, 2009.

6.2-8 Safety Evaluation by The Office of Nuclear Reactor Regulation Related to NRC Bulletin 96-03, Boiling Water Reactor Owners Group Topical Report NEDO-32686, "Utility Resolution Guidance for ECCS Suction Strainer Blockage," August 20, 1998, NUDOCS Accession No. 9809100159.

6.2-9 NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation, Enclosure 2 to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02," March 28, 2008, ADAMS Accession No. ML080230462.

6.2-10 NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Chemical Effects Evaluations, Enclosure 2 to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02," March 28, 2008, ADAMS Accession No. ML080380214.

6.2-11 WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," February 28, 2006, ADAMS Accession No. ML060890501.

6.2-12 C. B. Bahn, K.E. Kasza, W. J. Shack, and K. Natesean, Aluminum Solubility in Boron Containing Solutions as a Function of pH and Temperature, Argonne National Laboratory Contract Report to the NRC, September 19, 2008, ADAMS Accession No. ML091610696.

6.2-13 Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report (TR) WCAP-16406-P, Revision 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," December 27, 2007, ADAMS Accession No. ML073520295

[redacted]

May 12, 2010

Mr. Scott Head, Manager
Regulatory Affairs
STP Nuclear Operating Company
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SAFETY EVALUATION REPORT WITH OPEN ITEMS FOR CHAPTER 6 REGARDING THE SOUTH TEXAS PROJECT COMBINED LICENSE APPLICATION REVIEW

Dear Mr. Head:

The U.S. Nuclear Regulatory Commission staff is preparing a safety evaluation report (SER) with open items (OIs) for each chapter of the South Texas Project Units 3 & 4 Combined License Application (COLA) submitted by STP Nuclear Operating Company (STPNOC) on September 20, 2007.

The staff's SER with OIs for Chapter 6 will be provided to the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee to support upcoming meetings of the ACRS Subcommittee, scheduled to be held in June of 2010. The staff is continuing to review the COLA and may identify additional open items as a result of future STPNOC submittals.

In response to your letter dated April 15, 2010, the staff has removed the information identified as proprietary from the Chapter 6 SER with OIs. The enclosed SER with OIs is being provided again to STPNOC for review of proprietary information. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, we have determined that the enclosed SERs with OIs may contain proprietary information or other categories of information that should be withheld from public disclosure. We will delay placing the enclosures in the public document room to provide you with the opportunity to comment on information in the enclosed SERs with OIs that should be withheld from public disclosure.

NOTICE: Document transmitted herewith contains sensitive unclassified information. When separated from the enclosure this cover letter is "DECONTROLLED."

If you believe that any information in the enclosures is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390.

Sincerely,

/RA/

Mark Tonacci, Chief
ESBWR/ABWR Projects Branch 2
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-012
52-013

Enclosures:
Safety Evaluation Report with Open Items - Chapter 6

cc w/o encls: See next page

If you believe that any information in the enclosures is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390.

Sincerely,

/RA/

Mark Tonacci, Chief
ESBWR/ABWR Projects Branch 2
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-012
52-013

Enclosures:
Safety Evaluation Report with Open Items - Chapter 6

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