2.2 Departures from the Generic Technical Specifications

The following Tier 2 departures require prior NRC approval due to the changes to the reference ABWR Technical Specifications per 10 CFR 52 Appendix A Section VIII. C.4.

These departures are organized into three groups. The first are those Tier 2 design changes requiring implementing changes to the Technical Specifications. These are in Section 2.2.1.

The second group includes those changes to the Technical Specifications that change the intent but do not have a Tier 2 design departure as an underlying cause. These are in Section 2.2.2.

The third group (Section 2.2.3) are those changes to the wording of the generic Technical Specifications that do not change the intent and are not associated with a design change. These generally meet the definition of administrative departures used elsewhere in the COLA but require prior NRC approval since they change Technical Specifications.

2.2.1 Changes to the Technical Specifications due to Tier 2 Design Departures

STD DEP In all departures in this section, the change is authorized by law, will not represent an undue risk to the public health and safety, and is consistent with the common defense and security. Specific special circumstances unique to each departure are discussed as they apply.

STD DEP 6.2-2, Containment Analysis

Description

This departure updates the containment analysis for the ABWR DCD in ~~two~~three areas: (1) the modeling of flow and enthalpy into the drywell for the feedwater following a FWLB, ~~and~~ (2) the modeling of the drywell connecting vents for the FWLB and MSLB, and (3) the modeling of decay heat. A more detailed description is shown below.

This departure also makes the following changes: (1) it updates the suppression pool temperature limit from the DCD specified value of 97.2°C to a value of 100°C, and (2) it 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 for the FWLB, the maximum possible feedwater flow rate was calculated to be 164% of nuclear boiler rated (NBR) flow, based on the response of the feedwater pumps to an instantaneous loss of discharge pressure. Since the Feedwater Control System would respond to the decreasing reactor pressure vessel (RPV) water level by demanding increased feedwater flow, and there was no FWLB logic/mitigation in the certified ABWR design, this maximum feedwater flow was assumed to continue for 120 seconds. This was based on the following assumptions:

(1) All feedwater system flow is assumed to go directly to the drywell.

(2) Flashing in the broken feedwater line was ignored.

(3) Initial feedwater flow was assumed to be 105% NBR.

(4) The feedwater pump discharge flow will coast down as the feedwater system pumps trip due to low suction pressure. During the inventory depletion period, the flow rate is less than 164% because of the highly subcooled blowdown. A feedwater line length of 100 meters was assumed on the feedwater system side.

Subsequent to certification, analysis for plant-specific ABWRs revealed that these assumptions were non-conservative.

For the containment analysis, the feedwater system side of the FWLB has been changed using a revised time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy have been determined from the predicted characteristics of a typical feedwater system. The conservatism of the assumed mass flow and enthalpies will be confirmed after detailed condensate and feedwater designs are complete. In addition, to provide added assurance of acceptable results, safety related FWLB mitigation has been added to the STP 3 & 4 ABWR design which adds safety related instrumentation to sense and confirm a FWLB based on high differential pressure between feedwater lines coincident with high drywell pressure to trip the condensate pumps (Ref. STD DEP T1 2.4-2). This automated condensate pump trip is not credited in the containment analysis.

The analysis is further revised to reflect the characteristics of the horizontal vents configuration that had not been modeled in the DCD. The certified DCD model did not properly simulate the horizontal vent portion of the vent system and incorrectly modeled the vent clearing time. The revised STP 3 & 4 ABWR containment analysis has been performed using the drywell connecting vent (DCV) loss coefficients and considering the horizontal vents. The total DCV loss coefficient is based on a summation of losses.

Further analysis done based on ANSI/ANS-5.1 (1994), including the 2-sigma uncertainty, has determined that the decay heat curves used in the DCD based on best estimate ANSI/ANS-5.1 (1979) were non-conservative for long-term analysis. To address this, the decay heat curves used in the revised containment analysis were revised to reflect the ANSI/ANS-5.1 (1979) with 2-sigma uncertainty included.

The revised containment analysis uses the GOTHIC code and is documented in WCAP-17058. The analysis uses the same assumptions and inputs that were used in the DCD with consideration of the revised modeling as noted above. The report describes all input assumptions, baselining of the GOTHIC code results to those used in the DCD and all containment time-dependent pressure and temperature results. The report also evaluates the impact on the analysis results of the few unavoidable modeling differences due to certain features in the GOTHIC code.

The impact of the revised pressure and temperature results on pool swell velocity and height described in Appendix 3B is evaluated in a new departure which is STD DEP 3B-2.

Technical Specification 3.6.1.1, 3.6.1.2, and 3.6.1.4 Bases (Applicable Safety Analyses) are changed based upon the containment analysis. These changes show the peak containment pressure (P_a) from the containment analysis.

Evaluation Summary

This departure which updates the containment analysis for STP 3 & 4 does not affect Tier 1, Tier 2*, or any operational requirements. However, it does affect the Bases for Technical Specifications 3.6.1.1, 3.6.1.2, and 3.6.1.4 ~~and 3.6.1.6~~ and therefore requires NRC approval.

There is no impact on environmental qualification of equipment due to the higher predicted drywell temperatures and pressures. The qualification of equipment is based on the containment design pressures and temperatures. The calculated containment pressure and temperature for both the FWLB and MSLB remain below the design values except for a two second period when the drywell temperature exceeds the design temperature by 2.1°C for the MSLB. Due to thermal inertia, components in the drywell would not have sufficient time to reach the design limit temperature in such a short time.

The change in the design suppression pool temperature limit is to align the limit with the NPSH calculation assumptions to determine that adequate NPSH exists for the ECCS pumps. These calculations represent the limiting condition for determining maximum allowable suppression pool temperature. These calculations use a suppression pool temperature of 100°C. Therefore, the allowable design limit for the peak suppression pool temperature is being changed to 100°C.

The change in the assumed elapsed time between start of LOCA and initiation of suppression pool cooling and containment sprays from 10 minutes to 30 minutes is conservative relative to the DCD as heat removal from these systems is not credited until later in the accident sequence. This also brings the assumption for operator action into alignment with current safety analysis practices.

This departure was evaluated per Section VIII.C.4 of Appendix A to 10CFR part 52 and:

(1) This exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. The design change and revised containment analysis represents an improvement and therefore will not present an undue risk to the public health and safety. The change does not relate to security and does not otherwise pertain to the common defense and security.

(2) Special circumstance (iv) applies in that this represents a benefit in public health and safety. The ~~more advanced and complete analysis methods~~incorporation of these modeling changes as well as the use of an analysis method which has been baselined to the certified DCD analysis method provide a more accurate prediction of peak containment conditions post-accident. These results show that the peak containment pressure and temperature conditions calculated following an accident based on these improved analyses are ~~below the design limits~~acceptable.The FWLB mitigation to the ABWR design will provide added assurance that the revised containment analysis results will remain conservative when detailed feedwater and condensate system design and procurement work is completed.

As discussed above, the change satisfies the exemption criteria per the requirements in 10 CFR 52 Appendix A Section VIII.C.4.

STD DEP 7.3-7, Automatic Depressurization Subsystem (ADS) Manual Operation

This departure summary is relocated to Part 7, Section 3.0.

STD DEP 7.3-12, Leak Detection and Isolation System Sump Monitoring

Description

Subsection 7.3.1.1.2(m) of the reference ABWR DCD provides alarm setpoints (nominal values) to support Technical Specification limits for Reactor Coolant Pressure Boundary Leakage. The leakage rate values are also discussed in Subsections 5.2.5.4.1, 5.2.5.5.1, 5.2.5.5.2 and 5.2.5.9. The original values were based on a leak-before-break option (not used on STP 3 & 4) that allowed the use of a lower unidentified leakage limit and the removal of the unidentified leakage increase. In lieu of providing a plant-specific Leak Before Break analysis, drywell leakage rate limits are provided as follows:

- Total leakage averaged over the previous 24-hour period is changed from 95 L/min to 114 L/min
- Unidentified leakage is changed from 3.785 L/min to 19 L/min
- Unidentified leakage increase of 8 L/min within the previous 4-hour period in Mode 1 is added.

The 8 L/min increase in 4 hours is a plant computer based control room alarm that will provide an early warning to control room operators so they can take action well below the Technical Specification limit for unidentified leakage of 19 L/min. This alarm initiates on an increase in leakage above normal leakage values.

Technical Specification 3.4.3 (LCO, Actions B.1 and B.2, SR 3.4.3.1) and its associated Bases (Applicable Safety Analysis, LCO, Actions B.1 and B.2) are changed to show the new leakage values and the addition of an “increase in unidentified leakage” parameter.

Evaluation Summary

This departure changes the Technical Specification 3.4.3 LCO, SR, and associated Bases, but does not change the intent of the generic Technical Specification. However since it affects a Technical Specification it requires prior NRC approval. This departure was evaluated per Section VIII.C.4 of Appendix A to 10CFR part 52 and:

Special circumstance (iv) of 10 CFR 50.12(a)(2) applies in that substituting the operationally proven leakage limits was judged to be more conservative than applying the Leak Before Break criteria so that a net benefit to public health and safety results.

This exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

STD DEP 7.3-17, Automatic Depressurization System (ADS) Electrical Interface

Description

Subsection 7.3.2.1.2 (3e) of the reference ABWR DCD describes compliance with RG 1.75. The following information has been added to provide a more complete description of ECCS compliance with this RG:

“Sensor input signals are in Division I, II, III and IV. Control logic is performed in Divisions I, II and III.”

These words are added to clarify that control logic is only in Div I, II and III to conform to the three divisions of ECCS. However, sensor signals come from all four electrical Divisions.

Technical Specification 3.3.1.4 Bases (Background) is changed to show that there are three divisions of ESF logic (Divisions I, II, and III), not four.

Evaluation Summary

This departure changes Technical Specification 3.3.1.4 Bases but does not change the intent of the generic Technical Specification. However, since it is a change to the Technical Specification it requires prior NRC approval. This departure was evaluated per Section VIII.C.4 of Appendix A to 10CFR part 52 and:

Special circumstances are present as specified in 10 CFR 50.12(a)(2). Special circumstance (ii) applies in that the departure represents no change in the underlying purpose of the design but clarifies the Technical Specifications and conforms them to the standard design.

This exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the change represents an improvement to the Technical Specification and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security. As demonstrated above, the exemption complies with the requirements in Section VIII.C.4.

STD DEP 7.5-1, Post-Accident Monitoring (Drywell Pressure)

Description

The following changes are to assure the system designs meet the post-accident monitoring (PAM) design requirements of RG 1.97.

- Add variable Type A classification to the Drywell Pressure in Subsection 7.5.2.1. This information is to be used by the control room operator in determining an initiation of the drywell spray to maintain the reinforced concrete containment vessel below temperature limits under a LOCA condition. Table 7.5-3 has also been revised to add drywell pressure as a Type A variable.
- Correct the variable ranges for the Drywell Pressure and Meteorological Data in Table 7.5-2.
- Delete the Secondary Containment Air Temperature from the list of PAM variables in Table 7.5-2.
- Add Suppression Chamber (Wetwell) Spray Flow, variable Type D. Category 2, to the list of PAM variables in Table 7.5-2.
- Add the Type A wetwell pressure parameter to the large display panel in Subsection 18.4.2.11 to be used by the control room operator in determining a manual initiation of the wetwell spray to limit the bypass leakage.
- Add additional event items to Table 7.5-4 to align it with DCD Table 15A-9.

TS Bases B3.3.6.1 (LCO discussion for Function 5.a) is changed to show that Drywell Pressure and Wetwell Pressure are Type A Instruments (post-accident monitoring variables), and to show that Wetwell Atmosphere Temperature is a Category I variable and is a required function for post-accident monitoring. The Bases are also changed to provide a discussion for Wetwell Atmosphere Temperature (Function 13) and to show the correct reference to Wetwell Pressure, rather than Containment Wide Range Pressure.

TS Table 3.3.6.1-1 is changed so that Function 5.b shows Wetwell Pressure instead of Containment Wide Range Pressure, and Wetwell Atmosphere Temperature is added to that Table as Function 13.

Evaluation Summary

The PAM design requirements were updated to more closely follow the guidance of RG 1.97, BTP HICB-10 and TMI-related criteria 10 CFR 50.34. The identified changes are requirements that were exempted in the reference ABWR DCD but are now redesigned to better comply with RG 1.97.

From RG 1.97, Type A variables provide the primary information required to permit the control room operator to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis accident events. Based on that, the drywell and wetwell pressure readings are categorized as Type A variables as they are used by the operator to determine whether to initiate drywell or wetwell sprays to protect primary containment from exceeding pressure or temperature limits. Secondary containment air temperature is deleted as a PAM variable as it is not a required PAM variable in Table 2 of RG 1.97 or in 10 CFR 50.34.

Special circumstance (iv) applies in that the departure represents a net benefit to the public health and safety by providing a set of instruments for post-accident monitoring that are more closely in adherence with regulatory guidance. These include additional areas monitored for radiation, high profile display of containment conditions important to post-accident response, and modified instrument ranges to better bracket expected parameters post-accident.

This exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

STD DEP 7.7-10, Control Rod Drive Control System Interfaces

Description

Subsection 7.7.1.2.1 of the reference ABWR DCD provides the Rod Control and Information System (RCIS) interfaces with the Control Rod Drive (CRD) Control System for Single Rod Movement (Subsection 7.7.1.2.1(1)), Withdrawal Cycle (Subsection 7.7.1.2.1(2)), Insert Cycle (Subsection 7.7.1.2.1(3)), and Ganged Rod Motion (Subsection 7.7.1.2.1(4)). This COLA change implements the following revisions in the listed DCD subsections:

- The Performance and Monitoring Control System (PMCS) normal operational manual mode CRT display is replaced with the RCIS Dedicated Operator Interface on the Main Control Room Panel.
- The description of allowed operator single and ganged rod movement manual commands is revised.
- A discussion of the Rod Action and Position Information (RAPI) rod block operations is added.
- The description of RAPI normal rod movement operations is revised. The revised text includes description of operation of the Rod Server Modules (RSMs), the Rod Brake Controllers (RBCs), the Synchro-to-Digital Converters (SDCs), and the Fine Motion Control Rod Drives (FMCRDs).

- The descriptions of the CRD Control System Withdrawal Cycle and Insert Cycle interfaces are deleted.
- The description of ganged rod motion interface is revised.

Subsection B 3.9.4, “Bases,” of Section 16.0, “Technical Specifications” of ABWR DCD, Rev. 0, describes “Actions” to take to bypass an inoperable full-in position indication channel of the RCIS to allow refueling operations to proceed. The actions include identifying an alternate method to use to ensure the affected control rod is fully inserted. A possible option is described which uses the synchros of the affected FMCRD to verify rod full-in position. This COLA change implements a revision to the specific method described in Subsection B 3.9.4 to use the synchros to verify rod full-in position.

Evaluation Summary

This proposed change does not affect Tier 1 or Tier 2* information, however it does affect the bases of Technical Specification ~~3.4.3~~3.9.4 and therefore requires prior NRC approval. This proposed change was evaluated per Section VIII.C.4 of Appendix A to 10CFR part 52 and:

(1) This proposed change is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. This proposed change consists only of rewording for clarification and editorial correction with no change to the meaning or intent of the original bases of Technical Specification ~~3.4.3~~3.9.4, and therefore will not present an undue risk to the public health and safety. This proposed change does not pertain to the common defense and security.

(2) Special circumstances are present as specified in 10 CFR 50.12 (a) (2). Special circumstance (ii) applies to this exemption in that the application of the generic Technical Specifications without this change is not necessary to serve their underlying purpose.

As demonstrated above, the exemption complies with the requirements in Section VIII.C.4.

STD DEP 7.7-18, Rod Control and Information System Operator Information

Description

Subsection 7.7.1.2.3 of the reference ABWR DCD provides the Rod Control and Information System (RCIS) Reactor Operator Information. This COLA change implements the following revisions in Subsection 7.7.1.2.3:

- The list of RCIS annunciation activations at the main control panel is revised to be consistent with the current RCIS design.

- The list of RCIS status information provided on the RCIS dedicated operators interface on the main control panel is revised to be consistent with the current RCIS design.
- The list of RCIS operator functions allowed through the RCIS dedicated operators interface panel, and related RCIS displays, indications and associated controls provided on the main control room panel and on the RCIS cabinets and panels, is revised to be consistent with the current RCIS design.
- The list of RCIS related information displayed for the operator by main control room equipment other than the RCIS dedicated operator interface is revised to be consistent with the current RCIS design.

Bases Subsections B 3.9.3, “Background”, B 3.10.3, “Background” and “Applicable Safety Analyses”, B 3.10.4, “Background” and “Applicable Safety Analyses”, and B 3.10.5, “Background” and “Applicable Safety Analyses” of Section 16.0, “Technical Specifications” of the DCD reference use of an RCIS “Rod Test Switch” to allow two control rods to be withdrawn for scram testing. This “Rod Test Switch” is placed in Scram Test Mode through use of the RCIS Dedicated Operators Interface (DOI) panel (i.e., the switch is now a touch panel button). This COLA change implements a revision to Subsections B 3.9.3, B 3.10.3, B 3.10.4 and B 3.10.5 of the Technical Specifications to revise the manner in which the RCIS is placed in Scram Test Mode.

Evaluation Summary

This proposed change does not affect Tier 1 or Tier 2* information, however it does affect the bases of Technical Specifications 3.9.3, 3.10.3, 3.10.4 and 3.10.5 and therefore requires prior NRC approval. This proposed change was evaluated per Section VIII.C.4 of Appendix A to 10CFR part 52 and:

(1) This proposed change is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. This proposed change consists only of rewording for clarification and editorial correction with no change to the meaning or intent of the original bases of Technical Specifications 3.9.3, 3.10.3, 3.10.4 and 3.10.5 and therefore will not present an undue risk to the public health and safety. This proposed change does not pertain to the common defense and security.

(2) Special circumstances are present as specified in 10 CFR 50.12 (a) (2) which states, “Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

The description of the manner in which the RCIS is placed in Scram Test Mode requires revision in order to be consistent with the current design. Application of the regulation as stated in the Bases for Technical Specifications 3.9.3, 3.10.3, 3.10.4 and 3.10.5 is not necessary to serve the underlying purpose of the rule because the proposed change consists only of rewording for clarification and editorial correction with no change to the meaning or intent of the original bases of Technical Specifications.

As demonstrated above, the exemption complies with the requirements in Section VIII.C.4.

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

Description

The ABWR DCD provided a single 6.9 kV electrical system. This departure changes the medium-voltage electrical distribution system to a dual voltage system consisting of 13.8 kV and 4.16 kV. This departure will change the following:

- Medium voltage rating of the Power Generation (PG) buses increased to 13.8 kV,
- Medium voltage rating of the Plant Investment Protection (PIP) buses decreased to 4.16 kV,
- Medium voltage rating of the Class 1E buses decreased to 4.16 kV,
- EDG ratings increased to 7200 kW and 4.16 kV,
- Combustion turbine generator (CTG) ratings increased to 13.8 kV and at least 20 MWe,
- Time required for CTG to start and achieve steady state voltage and frequency increased from two minutes to “less than 10 minutes” as required by RG 1.155 for a Station Blackout (SBO) alternate AC source.

The 13.8 kV PG buses are nonsafety-related while the three emergency diesel generators provide power to divisional 4.16 kV safety buses for a more typical US practice. The change is necessary to allow the plant electrical distribution system to be designed and built using commercially available equipment. This will allow higher voltage (13.8 kV) to supply pumps at a greater distance, which will reduce starting impact on voltage regulation. Stub buses to the 4.16 kV will be included to accommodate the NRC required “direct connect to transformers” and to provide transformer differential current protection. An additional reserve auxiliary transformer from off-site power will be included.

10 CFR 50.63 states that if the CTG (alternate AC source) is available to power the Class 1E buses within 10 minutes of the onset of an SBO, then no coping analysis is required. The change in CTG startup time from two to < 10 minutes does not affect plant safety in the station blackout event since the CTG is not required to provide immediate core inventory or reactor pressure control.

As a result of this design change, Technical Specifications 3.3.1.4 (Table 3.3.1.4-1), and Bases 3.3.1.1 (Background) and 3.3.1.4 (Applicable Safety Analysis, LCO, and Applicability, Actions) are changed to show the medium voltage is 4.16 kV, not 6.9 kV.

Technical Specifications 3.5.1 (Actions), 3.8.1 (Actions, Surveillance Requirements), 3.8.4 (Actions), 3.8.9 (Actions), 3.8.11 (Actions) are changed to show electrical operating requirements changes for the CTG and DGs.

Technical Specifications Bases 3.8.1 (Background, Applicable Safety Analysis, LCO, Actions, Surveillance Requirements), 3.8.2 (LCO), 3.8.7 (LCO), 3.8.8 (LCO), 3.8.9 (Background, LCO, Table B 3.8.9-1), 3.8.11 (LCO, Actions), are changed to show electrical operating requirements changes for the CTG and DGs.

Evaluation Summary

This departure does not affect any Tier 1 or Tier 2* DCD. This design change results in changes to the Technical Specifications and their Bases as listed above but does not change the intent of the generic Technical Specifications. However since it affects Technical Specifications it requires prior NRC approval. This departure was evaluated per Section VIII.C.4 of Appendix A to 10CFR part 52 and:

(1) This departure is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. The departure will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

(2) Special circumstances are present as specified in 10 CFR 50.12(a)(2).

Special circumstance (iv) applies in that the departure represents a net benefit to the public health and safety by:

- The proposed design improves reliability with divisional and safety/non-safety isolation and independence increased through the use of stub buses
- The dual voltage design will provide greater flexibility for maintenance, surveillance, and inspection
- Increased availability of multiple sources of power to the various buses in the proposed design and the ability to isolate the buses individually, if needed.

As discussed above, the exemption complies with the requirements in Section VIII.C.4; therefore, STP requests approval.

STP DEP 8.3-3, Electrical Site Specific Power and Other Changes

Description

The associated standard departure STD DEP 8.3-1 revised the medium voltage electrical distribution system. Site specific changes per this departure are required to accommodate the new arrangements and electrical loads. These changes include diesel generator loading and other drawing changes listed below.

- Table 8.3-1 was updated to identify the site-specific changes (i.e., CT Fans, UHS, HECW Refrigerators, MCCs) as a result of performing load study calculations, diesel generator sizing and CTG sizing calculations.
- Figure 8.3-1, Sheets 1-4, were revised to incorporate site-specific load changes which were identified during the process of performing load study calculations, diesel generator sizing and CTG sizing calculations.

As a result of this design change, Technical Specifications Bases 3.8.9 (Table B 3.8.9-1 “AC, DC, and AC Vital Bus Electrical Power Distribution System”) is changed to show the AC Bus changes, i.e., MCC changes.

Evaluation Summary

This departure does not affect any Tier 1 or Tier 2* information. This design change results in changes to the Technical Specification Bases as listed above but does not change the intent of the generic Technical Specifications. However since it affects Technical Specifications it requires prior NRC approval. This departure was evaluated per Section VIII.C.4 of Appendix A to 10CFR part 52 and:

(1) This departure is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. This departure will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

(2) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Special circumstance (ii) applies to this exemption in that application of the Technical Specification Bases information from the reference ABWR DCD is not necessary to serve the underlying purpose of the rule as follows:

(a) This departure revises EDG loads within allowable total capacity and changes electrical bus arrangements to support site-specific features.

(b) The load study calculations for the site-specific loads on the EDGs and CTG for STP 3&4 have been performed and the results validate the loads included in updated Table 8.3-1 and Figure 8.3-1 and show that the EDGs retain sufficient operating margin to meet NRC regulatory position in RG 1.09, Rev. 4.

As discussed above, the exemption complies with the requirements in Section VIII.C.4; therefore, STP requests approval.

STD DEP 10.4-5, Condensate and Feedwater System

Description (Technical Specification Affected)

Technical Specification 3.3.4.2 Bases (Background) is changed to show that there are four feedwater pumps which requires four feedwater pump Adjustable Speed Drives (ASDs). The reference ABWR DCD specified two feedwater pump ASDs.

Evaluation Summary (Technical Specification Affected)

Special circumstance as defined in 10CFR50.12 (ii) applies in that the original DCD design is not necessary to achieve the underlying purpose of the rule. The departure represents an equal or better alternative in that feedwater flows are controlled via variable speed drives, reducing energy loss at part throttle, additional redundancy exists with additional spare pumps, and critical components like demineralizers can operate at lower pressure.

This exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

The remaining changes associated with this departure are described and evaluated in the following sections.

Description (Technical Specification Not Affected)

The reference ABWR DCD and the updated plant configuration based on departures from the reference ABWR DCD are compared in the table below:

Reference ABWR DCD	Reference ABWR DCD with Departures
▪ Four condensate pumps	▪ Four condensate pumps
▪ Three reactor feed pumps	▪ Four condensate booster pumps
▪ Two heater drain pumps	▪ Four reactor feed pumps
▪ One or more heater drain tanks	▪ Four heater drain pumps
▪ A feed pump bypass valve controls FW during plant startup	▪ One high pressure heater drain tank
▪ Two offgas recombiner condensers cooled by the condensate system	▪ Three low pressure heater drain tanks
	▪ One low flow control valve in feed pump discharge header for startup
	▪ One bypass valve used for bypassing HP heaters
	▪ Two offgas recombiner condensers cooled by the turbine building cooling water system

The addition of condensate booster pumps eliminates the necessity to design condensate pumps with high discharge head and the necessity to design the equipment downstream of the condensate pumps (filter/demineralizers, auxiliary equipment coolers) for high pressure application. The addition of the booster pumps allows the design of the condensate pumps to have low discharge head. The condensate and condensate booster pumps provide the necessary NPSH to the reactor feed pump suction.

The addition of one reactor feed pump and two heater drain pumps improves plant availability. If one of these pumps trip during normal operation, the standby pump starts automatically to maintain rated power operation. The use of four booster pumps allow three to be in operation, and the automatic startup of the standby pump as needed to support full power operation.

The four heater drain pumps take suction from one common heater drain tank, which collects drains from the high pressure feedwater heaters and the moisture separator drain tanks. The use of one heater drain tank is based upon equipment arrangement consideration in the Turbine Building.

Three low-pressure drain tanks have been added and each drain tank is located downstream of the corresponding Nos. 1 and 2 heaters. For example, low pressure drain tank A is common to 1A and 2A heaters. This configuration eliminates the condition of drain choking if the differential pressure between No. 1 heater drain cooler and the main condenser becomes insufficient.

The offgas recombiner condensers are changed from being cooled by condensate to being cooled by the turbine building cooling water system to ensure the operating temperature does not exceed the limit for the offgas system.

One low-flow control valve is added to meet the low flow condition during plant startup. A bypass valve used for bypassing the high pressure feedwater heaters is added to provide an additional operation mode when one high pressure heater is not in operation.

Evaluation Summary (Technical Specification Not Affected)

The changes described above are limited to the equipment addition and removal of non-safety-related equipment, and the changes do not affect any safety-related SSC, or SSC important to safety. Therefore, the changes described in this departure other than those specifically described above as affecting the Technical Specifications do not affect the safety analyses , have no impact on any safety function or any SSC used to mitigate the consequences of an accident, and do not increase the consequences of a malfunction of an SSC important to safety. Furthermore, there is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety, and the departure will not result in any accident of a different type than previously evaluated in the referenced DCD.

The changes described in this departure other than those specifically described above as affecting the Technical Specifications have been evaluated pursuant to the requirements in 10CFR52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1 or Tier 2* DCD, Technical Specifications, Bases for Technical Specifications or operational requirements as a result of these changes.

STD DEP 16.3-39, LCO 3.3.4.2, Feedwater and Main Turbine Trip Instrumentation

This departure modifies the instrumentation description and required actions for TS 3.3.4.2 and the associated Bases. The description in LCO 3.3.4.2 has been clarified to state that the three Feedwater Pump and Main Turbine Trip Instrumentation channels consist of three instrumentation channels and three digital controllers. The Entry Condition and Required Actions in the LCO have been modified accordingly to reflect both the instrumentation channels and the digital controllers. Required actions to place an inoperable channel in trip or bypass have been deleted. The Surveillance Requirement (SR) 3.3.4.2.2 has been revised to add a note permitting delayed entry into the associated Conditions and Required Actions when performing functional testing. The Bases have been revised to reflect these changes.

2.2.2 STD DEP Changes of Intent to the Technical Specifications

The following departures change the wording and the intent of the referenced ABWR DCD Technical Specifications. None of these changes is caused by nor are the departures related to changes to an underlying design.

STD DEP 16.2-1, Safety Limit Violation

Description

The following Generic Technical Specifications and the associated Bases related to Safety Limits have been deleted:

- Specification 2.2.1 requiring NRC notification within 1 hour of any Safety Limit violation;
- Specification 2.2.3 requiring notification of the General Manager - Nuclear Plant, Vice President - Operations, and Offsite Reviewers as specified in Specification 5.5.2 within 24 hours of a violation;
- Specification 2.2.4 requiring the submittal of a Licensee Event Report within 30 days to the NRC of a Safety Limit Violation; and,
- Specification 2.2.5 requiring NRC authorization to resume unit operation.

Specifications 2.2.1, 2.2.4 and 2.2.5 are duplicative of the requirements found in 10 CFR 50.72, 10 CFR 50.73, and 10 CFR 50.36(d)(1), respectively and thus do not belong in Technical Specifications. Specification 2.2.3 does not meet the criteria for inclusion in Technical Specifications and is being relocated to a conduct of operations-type procedure developed in accordance with the procedures development plan.

Evaluation Summary

This departure was evaluated per Section VIII.C.4 of Appendix A to 10 CFR Part 52, which requires that 1) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2); As shown below, both of these two criteria are satisfied.

- The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the actions in question are required by regulations or plant documents and therefore the deletions of the actions from the Technical Specifications will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

- Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since it is unnecessary to include the deleted provisions in the Technical Specifications in order to ensure that the actions are accomplished.

As demonstrated above, this exemption complies with the requirements in Section VIII.C.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

References

(1) 10 CFR 50.36, Technical Specifications

(2) 10 CFR 50.12, Specific Exemptions

STD DEP 16.3-39, LCO 3.3.4.2, Feedwater and Main Turbine Trip Instrumentation

Description

This departure modifies the instrumentation description and required actions for TS 3.3.4.2 and the associated Bases. The description in LCO 3.3.4.2 has been clarified to state that the three Feedwater Pump and Main Turbine Trip Instrumentation channels consist of three instrumentation channels and three digital controllers. The Entry Condition and Required Actions in the LCO have been modified accordingly to reflect both the instrumentation channels and the digital controllers. Required actions to place an inoperable channel in trip or bypass have been deleted. The Surveillance Requirement (SR) 3.3.4.2.2 has been revised to add a note permitting delayed entry into the associated Conditions and Required Actions when performing functional testing. The Bases have been revised to reflect these changes.

Evaluation Summary

This departure was evaluated per Section VIII.C.4 of Appendix A to 10 CFR Part 52, which requires that 1) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2); As shown below, both of these two criteria are satisfied.

- The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, these changes are made to be consistent with the actual design of the ABWR instruments and do not change the underlying design. This will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

- Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since it is necessary for the Technical Specifications to reflect the certified design in Tier 1 and Tier 2.

As demonstrated above, this exemption complies with the requirements in Section VIII.C.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP 16.3-78, LCO 3.3.6.1, Post Accident Monitoring (PAM) Instrumentation

Description

The containment water level parameter has been removed from Post Accident Monitor technical specifications. The instrumentation does not meet the Bases' criteria for inclusion (i.e., Drywell water level is classified as Cat. 2 and sump level is classified as Cat 3). Also, the Bases only require that Post-Accident Monitoring instruments that are classified as Regulatory Guide 1.97 Type A or Category I be included. Lower drywell level instrumentation is described as “not warranted” in the DCD.

Evaluation Summary

This departure was evaluated per Section VIII.C.4 of Appendix A to 10 CFR Part 52, which requires that 1) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2); As shown below, both of these two criteria are satisfied.

- The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the Bases only require that Post-Accident Monitoring instruments that are classified as Type A or Category I be included and so will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

- Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since it is unnecessary to include the deleted provisions in the Technical Specifications in order to ensure that they reflect DCD design and regulatory guidance.

As demonstrated above, this exemption complies with the requirements in Section VIII.C.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP 16.5-1, Unit Responsibility

Description

Technical Specification 5.1.2 states: “During any absence of the [SS] Shift Supervisor/Manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] Shift Supervisor/Manager from the control room while the unit is in MODE 5, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.”

Technical Specification 5.1.2 is being changed as follows: “During any absence of the [SS] Shift Supervisor/Manager from the control room while the unit is in MODE 1, 2, or 3, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] Shift Supervisor/Manager from the control room while the unit is in MODE 4 or 5, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.”

10 CFR 50.54 (m) (2) (iii) states, “When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.”

MODE 4 is being deleted from the first statement and added to the second statement. MODE 4 is defined as cold shutdown and does not require an SRO to assume control room command function, but allows this individual to be an RO or SRO. This change is consistent with the requirements as stated in 10 CFR 50.54 (m) (2) (iii).

Evaluation Summary

This departure was evaluated per Section VIII.C.4 of Appendix A to 10 CFR Part 52, which requires that 1) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2). As shown below, both of these two criteria are satisfied.

(1) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

(2) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since the departure represents a clarification of an acceptable process of compliance with current regulatory requirements and therefore will result in a benefit to the public health and safety.

As demonstrated above, this exemption complies with the requirements in Section VIII.C.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

References

(1) 10 CFR 50.54 (m) (2) (iii)

STD DEP 16.5-2, Unit Staff

Technical Specification 5.2.2.a, “Unit Staff” states: “An auxiliary operator shall be assigned to each reactor containing fuel and an additional auxiliary operator shall be assigned for each control room from which a reactor is operating.” Technical Specification 5.2.2.a, Note 1, states: “Two unit sites with both units shutdown or defueled require a total of three auxiliary operators for the two units.” Technical Specification 5.2.2.d states: “Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators and key maintenance personnel).”

In all three instances in Technical Specification 5.2.2, “Unit staff,” the term “auxiliary operator” is changed to “non-licensed operator.” This administrative change is modifying the Technical Specification to be consistent with industry terminology.

STD DEP 16.5-3, Technical Specification Bases Control Program

Description

Technical Specification 5.4.2.b states: “A change to the site-specific portion of the FSAR that involves an unreviewed safety question as defined in 10 CFR 50.59, or a change to Tier 2 of the plant-specific DCD that involves an unreviewed safety question as defined in the design certification rule for the ABWR (Appendix A to 10 CFR 52).”

Technical Specification 5.4.2.b is being changed to: “A change to the site-specific portion of the FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59, or the design certification rule for the ABWR (Appendix A to 10 CFR 52).”

This change is being made to properly define the Technical Specification Bases Control Program process for operation of the units after the license is approved.

Evaluation Summary

This departure was evaluated per Section VIII.C.4 of Appendix A to 10 CFR Part 52, which requires that 1) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2). As shown below, both of these two criteria are satisfied.

(1) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the departure is a clarification of an acceptable process for compliance with current regulatory requirements and therefore will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

(2) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since change is to bring this section into agreement with 10 CFR 52.

As demonstrated above, this exemption complies with the requirements in Section VIII.C.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP 16.5-4, Reporting Requirements

Description

Technical Specification 5.7.1.1 states: “Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted by March 31 of each year. The initial report shall be submitted by March 31 of the year following initial criticality.”

Technical Specification 5.7.1.1 will be changed to state: “Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted by April 30 of each year. The initial report shall be submitted by April 30 of the year following initial criticality.”

10 CFR 20.2206 requires this due date by April 30 of each year, this change is necessary to be consistent with the CFR.

Evaluation Summary

This departure was evaluated per Section VIII.C.4 of Appendix A to 10 CFR Part 52, which requires that 1) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2). As shown below, both of these two criteria are satisfied.

(1) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the departure is a clarification of an acceptable process for compliance with current regulatory requirements and therefore will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

(2) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since this change is to bring this section into agreement with 10 CFR 52.

As demonstrated above, this exemption complies with the requirements in Section VIII.C.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

References

(1) 10 CFR 20.2206 Reports of Individual Monitoring

STD DEP 16.5-5, TS 5.2.2 Unit Staff - Working Hours

Description

The proposed change removes working hour limits imposed in the Technical Specifications in order to support compliance with 10 CFR Part 26, Subpart I, Managing Fatigue. Work hour controls and fatigue management requirements have been incorporated into the NRC's regulations; therefore, it is unnecessary to have work hour control requirements in the Technical Specifications.

Technical Specification 5.2.2.d requires administrative procedures to be developed and implemented to limit the working hours of personnel who perform safety related functions. This change proposes to eliminate these Technical Specification requirements because they are superseded by the requirements in 10 CFR Part 26, and to renumber paragraphs 5.2.2.e and 5.2.2.f accordingly.

On April 17, 2007, the NRC Commissioners approved a final rule amending Title 10, Part 26, of the Code of Federal Regulations (CFR) which, among other changes, established requirements for managing worker fatigue at operating nuclear power plants. Subpart I specifically addresses managing worker fatigue by designating individual break requirements, work hour limits, and annual reporting requirements. Subpart I was published in the Federal Register on March 31, 2008 (73 FR 16966) and Notice of Availability was given in the Federal Register on December 30, 2008 (73 FR 79923) to remove this subsection from Technical Specifications through TSTF-511, “Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26.”

Evaluation Summary

This departure was evaluated per Section VIII.C.4 of Appendix A to 10 CFR Part 52, which requires that 1) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2).

As shown below, both of these two criteria are satisfied.

- The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the actions in question are required by regulations or plant documents and therefore the deletions of the actions from the Technical Specifications will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.

- Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since it is unnecessary to include the deleted provisions in the Technical Specifications in order to ensure that the actions are accomplished.

As demonstrated above, this exemption complies with the requirements in Section VIII.C.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

2.2.3 STD DEP Technical Specifications Editorial Revisions and Clarifications

The following departures change the Technical Specification wording but neither change nor are caused by an underlying design departure and do not change the intent of the generic Technical Specifications.

These departures are included here and not in Sections 2.2.1 or 2.2.2 above as they meet the general rules for administrative departures as defined in section 1.0. Since they affect Technical Specifications, they require prior NRC approval and are included in this section rather than Section 4.0.

These departures were evaluated per Section VIII.C.4 of Appendix A to 10 CFR Part 52, which requires that 1) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and 2) special circumstances are present as specified in 10 CFR 50.12(a)(2). As shown below, each of these criteria are satisfied.

(1) These exemptions are not inconsistent with the Atomic Energy Act or any other statute and therefore are authorized by law. As discussed above, the departures are administrative and therefore will not present an undue risk to the public health and safety, and the departures do not relate to security and do not otherwise pertain to the common defense and security.

(2) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Special circumstance (ii) applies to these exemptions in that the application of the generic Technical Specifications without these changes would not serve their underlying purpose.

As demonstrated above, these exemptions comply with the requirements in Section VIII.C.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve these exemptions.

STD DEP 16.2-2, Safety Limits

The BWR designs prior to the ABWR had reactor coolant pumps external to the reactor vessel. In the ABWR design the reactor coolant pumps are an internal design. The Technical Specification as written is based on external reactor coolant pumps. The limits for suction and discharge piping are being eliminated for the ABWR Specification to reflect the current design, since the pumps are internal, there are no external pump piping. As cited in DCD Section 5.4.1, all ten recirculation pumps are located inside the reactor coolant pressure boundary. Therefore, the safety limits on pressure for the suction and discharge piping are deleted.

During the design detailing stage of the ABWR development and DCD review, this change was noted and the Technical Specifications updated to reflect the current design. This change to the reference ABWR DCD Technical Specifications is intended to maintain consistency between the design description and the Technical Specifications.

STD DEP 16.3-1, 3.0, Limiting Condition for Operation (LCO) Applicability

LCO 3.0.6 references the Specification 5.8 for the Safety Function Determination Program. The actual Specification number for the Safety Function Determination Program is Specification 5.6. Therefore, the Specification number has been corrected in both the Specification and Bases.

STD DEP 16.3-2, LCO 3.0 and Surveillance Requirements (SRs)

The Bases for SR 3.0.1 state that the high pressure core flooder (HPCF) System requires a functional test to be performed at a specified reactor pressure. The HPCF System does not require reactor steam to operate because it utilizes electrical power. The statement is incorrect. The Reactor Core Isolation Cooling System is the appropriate system that should have been referenced. Therefore, high pressure core flooder has been replaced by the Reactor Core Isolation Cooling System.

The Bases for SR 3.0.1 also refers to control rod maintenance during refueling as an example for mode applicability of surveillance requirements. Two errors are corrected in this paragraph. First, the Bases for SR 3.0.1 refers to SR 3.1.3.4 for scram time testing. The appropriate SR is SR 3.1.4.3. SR 3.1.4.3 is the Surveillance performed at lower reactor pressures and is appropriate for this Bases discussion. Second, this Bases section also indicates that the control rod drive scram time testing should be performed at reactor steam dome pressures greater than 5.51 MPaG. However per Technical Specification 3.1.4 SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 this testing should be performed at reactor steam dome pressures greater than or equal to 6.55 MPaG. The pressure listed in the description for the Bases for SR 3.0.1 is revised accordingly.

STD DEP 16.3-3, LCO 3.1.7, Standby Liquid Control (SLC) System

The Bases states that “Because the minimum required boron solution concentration is the same for both ATWS mitigation and cold shutdown (unlike some previous reactor designs) then if the boron solution concentration is less than the required limit, both

SLC subsystems shall be declared inoperable.” The Required Actions for LCO 3.1.7 include Condition A that requires entry when concentration of boron in solution is not within limits. The Required Action is to restore concentration of boron in solution to be within limits within 72 hours and 10 days from discovery of failure to meet the LCO. The Bases is not consistent with the LCO therefore, the Bases statement has been deleted.

STD DEP 16.3-4, LCO 3.1.1, Shutdown Margin (SDM)

The Bases of SR 3.1.1.1 states, “Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the Rod Worth Minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, “Control Rod Testing-Operating”).” This statement has been replaced with, “This testing is performed in accordance with LCO 3.10.7, “Control Rod Testing-Operating” or LCO 3.10.8, “SDM Test-Refueling” where additional requirements are required to be met.” This change is made to be consistent with the Specifications in the Special Operations LCO section of the Technical Specifications.

LCO 3.10.7 states, “The requirements of LCO 3.1.6, “Rod Pattern Control,” may be suspended and control rods bypassed in the Rod Action and Position Information (RAPI) Subsystem as allowed by SR 3.3.5.1.7, to allow performance of SDM demonstrations, control rod scram time testing, control rod friction testing, and the Startup Test Program, provided conformance to the approved control rod sequence for the specified test is verified by a second licensed operator or other qualified member of the technical staff.” This has been replaced with the requirements of LCO 3.1.6 - “Rod Pattern Control” - may be suspended to allow performance of SDM demonstrations, control rod scram time testing, control rod friction testing, and the Startup Test Program, provided LCO 3.3.5.1 - “Control Rod Block Instrumentation” - for Function 1.b of Table 3.3.5.1-1 is met with the approved control rod sequence or conformance to the approved control rod sequence for the specified test is verified by a second licensed operator or other qualified member of the technical staff.” Consistent changes have also been made to the Bases and surveillances have been modified to ensure the LCO is met depending on the option taken. These changes are made to be consistent with the allowances in LCO 3.10.8 that either requires the LCO 3.3.5.1, “Control Rod Block Instrumentation,” MODE 2 requirements for Function 1.b of Table 3.3.5.1-1 or conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff.

STD DEP 16.3-5, LCO 3.4.1, Reactor Internal Pumps (RIPs)-Operating

The LCO 3.4.1 requires the reactor internal pumps to be “operating.” SR 3.4.1.1 requires the reactor internal pumps to be OPERABLE. SR 3.4.1.1 has been revised to be consistent with the LCO. Therefore, the SR is modified to require the pumps to be operating. The same change has been made to the Bases of SR 3.4.1.1.

STD DEP 16.3-6, LCO 3.4.1, Reactor Internal Pumps (RIPs)-Operating

The Bases Background section of LCO 3.4.1 states that, “The reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e. 55 to 100% RTP).” The lower end of the range has been changed from “55” to “70” to be consistent with the design (i.e., DCD Section 5.4.1.2 Power Generation Design Bases).

STD DEP 16.3-7, LCO 3.4.2, Safety/Relief Valves (S/RVs)

The Bases states “The transient evaluations in Reference 3 are based on these setpoints, but also include the additional uncertainties of +/- 1% of the nominal setpoint to account for potential setpoint drift to provide an added larger degree of conservatism.” Reference 3 is DCD Chapter 15. The transients in Chapter 15 do not take credit for the “safety” function of the safety relief valves, but credit the “relief” function. Therefore, the phrase has been modified to indicate that the “overpressurization evaluation” is the appropriate event as documented in Reference 2 (DCD Tier 2, Section 5.2.2).

STD DEP 16.3-8, LCO 3.4.9 RCS Pressure and Temperature (P/T) Limits

The LCO 3.4.9 Bases Applicable Safety Analyses of section states that, “Reference 7 establishes the methodology for determining the P/T limits.” Reference 7 is NEDO 21778-A, which is not the correct reference. This document does not include the methodology for determining P/T Limits for the ABWR. Therefore it has been deleted as a reference.

Additionally, the bracketed information in Section 5.7.1.6 has been replaced with "Regulatory Guide 1.99, Revision 2 and in accordance with 10 CFR 50, Appendix G." The methodology establishing the ABWR P/T Limits is described in Regulatory Guide 1.99 Revision 2 and is in accordance with 10 CFR 50, Appendix G, as described in the Bases Background section of LCO 3.4.9.

STD DEP 16.3-9, LCO 3.4.7 Alternate Decay Heat Removal

The Fuel Pool Cooling and Cleanup System can be used as an alternate source of decay heat removal in MODE 5. It cannot be used for decay heat removal in MODE 3 or MODE 4. The Bases required action for LCO 3.4.7 and LCO 3.4.8 states that the Spent Fuel Pool Cooling System may be used as an alternate decay heat removal system during MODE 3 and MODE 4, respectively. The Bases of LCO 3.9.7 and LCO 3.9.8 do not include the Spent Fuel Pool Cooling System as a method for alternate decay heat removal system during MODE 5 operations. The Fuel Pool Cooling and Cleanup System has been deleted as an alternate source of decay heat removal from Technical Specifications 3.4.7 and 3.4.9. Likewise, the Fuel Pool Cooling and Cleanup System has been added to the Bases for Action A1 in Sections 3.9.7 and 3.9.8. The name “Spent Fuel Pool Cooling System” is changed to “Fuel Pool Cooling and Cleanup (FPC) System” to be consistent with DCD Tier 1, Section 2.6.2 and the plant specific P&IDs.

STD DEP 16.3-10, LCO 3.5.1, ECCS-Operating

The Bases Background provides a range of pressures in which the High Pressure Core Flooder (HPCF) System and the Reactor Core Isolation Cooling System are designed to operate. The appropriate ranges for HPCF and RCIC have been included in the Bases.

The Bases Background states that HPCF System includes a full flow test line that routes water from and to the CST. CST has been replaced with suppression pool to be consistent with the actual design.

The Bases Background provides a description of the pneumatic supply to the Automatic Depressurization System valves. The Bases description has been modified for clarity.

The LCO Bases provide a summary of Specifications that support the Function of the Emergency Core Cooling Systems during the operating MODES. Specification LCO 3.7.2, RCW/RSW and UHS-Shutdown, and LCO 3.7.3, RCW/RSW and UHS-Refueling are also referenced in the Bases discussion. These Specification do not apply during the same MODES as Specification 3.5.1, therefore the Specifications have been deleted from the Bases discussion.

STD DEP 16.3-11, 3.4.3 RCS Operational LEAKAGE

The Bases states, “Crack behavior from experimental programs (Refs. 4 and 5) shows leak rates of tens of thousands liters per second will precede crack instability. The text change from “tens of thousands liters per second” to “hundreds of liters per minute” is consistent with industry guidance and NRC communications.”

Reference 5 is NUREG-76/067, October 1975. This reference has been changed to NUREG 75/067 since it is the appropriate reference number.

STD DEP 16.3-12, LCO 3.9.7, Residual Heat Removal Flow Path

The ABWR DCD Technical Specification (TS) Bases B 3.9.7 and B 3.9.8 describe the flow path of the RHR Shutdown Cooling System to the reactor pressure vessel. For RHR subsystems B and C it currently states that each pump discharge to the reactor is via the “RHR inlet nozzles.” This has been changed to “RHR low pressure flooder spargers” consistent with design and Bases B 3.4.7 and B 3.4.8.

STD DEP 16.3-13, LCO 3.9.8, Residual Heat Removal (RHR) - “Low Water Level” Applicability

The Applicability of Specification 3.9.8 is MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level < 7.0 m above the top of the RPV flange. The LCO Bases states, “In MODE 5 with the water level < 7.0 m above the reactor pressure vessel (RPV) flange two RHR shutdown cooling subsystems must be OPERABLE.” The Bases description has been modified to be consistent with the Specification.

In addition the Bases for the Required Actions state, “If at least one RHR subsystem is not restored to OPERABLE status immediately, additional actions are required to minimize any potential fission product release to the environment.” The sentence has been changed to “With the required shutdown cooling subsystem(s) inoperable and the required alternate method(s) of decay heat removal not available in accordance with Required Action A.1, additional actions are required to minimize any potential fission product release to the environment." This change is made to be consistent with the requirements in the Specification.

STD DEP 16.3-14, LCO 3.9.2, Refuel Position Rod-Out Interlock

The Applicability Bases refers to LCO 3.1.2, Reactivity Anomalies, when referring to control rods. The appropriate LCO is 3.1.3, Control Rod OPERABILITY.

STD DEP 16.3-15, LCO 3.9.5, Control Rod OPERABILITY - Refueling

The accumulator pressure in SR 3.9.5 has been changed from 10.49 to 12.75 MPAG to be consistent with the Bases. A lower pressure in the accumulators has not been determined for the ABWR design. This pressure is consistent with the pressure in LCO 3.1.5, Control Rod Scram Accumulators in MODE 1 and 2.

STD DEP 16.3-16, LCO 3.7.1, Reactor Building Cooling Water (RCW) System, Reactor Service Water (RSW) System, and Ultimate Heat Sink (UHS)-Operating, ~~and~~ LCO 3.7.2, ~~Reactor Building Cooling Water (~~RCW~~) System, Reactor Service Water (~~/RSW~~)~~ System and ~~Ultimate Heat Sink (~~UHS~~)~~ - Shutdown and LCO 3.7.3, RCW/RSW System and UHS - Refueling

LCO 3.7.1 includes a Required Action C.2 that requires restoration of two inoperable RCW/RSW or UHS divisions to OPERABLE status within 14 days. LCO 3.7.2 includes a Required Action B.2 that requires restoration of two inoperable RCW/RSW or UHS divisions to OPERABLE status within 14 days. These Required Actions have been deleted since redundant requirements are already included in Condition A of each Specification. The change is consistent with the Completion Time Rules of Section 1.3.

Likewise these actions, which are described in Technical Specification Bases sections B3.7.1 and B3.7.2, have been deleted.

The STP 3&4 UHS design incorporates cooling towers with fans and a UHS basin instead of a UHS spray pond. LCOs 3.7.1, 3.7.2, and 3.7.3 are revised to include SRs 3.7.1.4, 3.7.2.4, and 3.7.3.4, respectively, for monthly surveillance testing of the cooling tower cell fans.

STD DEP 16.3-17, LCO 3.10.12, Multiple Control Rod Drive Subassembly Removal - Refueling

LCO 3.10.12 states, "The requirements of LCO 3.9.3, "Control Rod Position"; LCO 3.9.4, "Control Rod Position Indication"; and LCO 3.9.5, "Control Rod OPERABILITY - Refueling," may be suspended, and the "full in" position indicators may be bypassed for any number of control rods in MODE 5, to allow removal of control rod drive subassemblies with the control rods maintained fully inserted by their anti-rotation devices."

SR 3.10.12.1 requires verification that the anti-rotation devices associated with each CRD subassembly removed are in the correct position to maintain the control rod fully inserted.

The word “applicable” has been added to the LCO and Surveillance just before the “anti-rotation devices” to indicate that not both of the anti-rotation devices are required to maintain the rod in the correct position to maintain the control rod fully inserted.

The Bases Background has been updated to describe when each device applies. Further information is provided in DCD Section 4.6.2.3.4, CRD Maintenance. In addition, Bases discussions for the Applicable Safety Analyses, and Applicability have been modified.

STD DEP 16.3-18, LCO 3.10.8 SHUTDOWN MARGIN (SDM) Test - Refueling

The LCO 3.10.8 Bases for ACTIONS “A.1” and “B.1” have been modified to more accurately reflect the LCO 3.10.8 ACTIONS. In particular, the phrase “for reasons other than Condition B” has been added to the Bases for ACTION A.1 to accurately reflect LCO 3.10.8 ACTION “A,” and the Bases for ACTION B.1 have been revised to more specifically describe the Condition as “one control rod not coupled to its associated CRD.” Additionally, the Bases for Action B.1 have been revised to clarify that upon completion of Action B, the LCO 3.9.5 requirements for an inoperable control rod apply.

STD DEP 16.3-19, LCO 3.10.4, Control Rod Withdrawal - Cold Shutdown

LCO 3.10.4 states “The reactor mode switch position specified in Table 1.1 1 for MODE 4 may be changed to include the refuel position, and operation considered not to be in MODE 2, to allow withdrawal of a single control rod or control rod pair, and subsequent removal of the associated control rod drives (CRD) if desired, provided the following requirements are met.”

LCO 3.10.4 part 2 states, “All other control rods in a five by five array centered on the control rod being withdrawn are disarmed.”

LCO 3.10.4 part 2 has been revised to indicate that the control rods that must be disarmed must include all other control rods in a five by five array centered on the control rod “or control rod pair” being withdrawn. This change is consistent with LCO 3.10.4 and SR 3.10.4.2.

STD DEP 16.3-20, LCO 3.10.4, Control Rod Withdrawal - Cold Shutdown

LCO Bases provides a list of other Special Operations LCO applicable in MODE 4 with the reactor mode switch in the refuel position. In this list, “LCO 3.10.3, Control Rod Withdrawal-Hot Shutdown,” is listed. This Specification is applicable in MODE 3 with the reactor mode switch in the refuel position. Reference to the Specification is deleted since it does not apply.

STD DEP 16.3-21, LCO 3.10.5, Control Rod Drive (CRD) Removal - Refueling

Technical Specification LCO 3.3.1.1, Functions 2.a, APRM Neutron Flux-High, Setdown and Function 2.d, APRM-Inop is applicable in MODES 2 and MODES 1 and 2, respectively. LCO 3.10.5 is applicable in MODE 5 with LCO 3.9.5 not met. LCO 3.10.5 allows the requirements of and Function 2.a and Function 2.d to not be met when in utilizing this Special Operations LCO. Since LCO 3.10.5 is used when in

MODE 5 with LCO 3.9.5 not met, there is no specific need to except the requirements of Function 2.a and 2.b. Therefore they have been deleted from the LCO statement.

STD DEP 16.3-23, LCO 3.10.5, Control Rod Drive (CRD) Removal - Refueling

The LCO Bases refer to LCO 3.3.8.2 instead of LCO 3.3.8.1. It also does not utilize the correct Specifications Titles for LCO 3.3.1.2 and 3.3.8.1. In addition, the Applicability Bases also refers to LCO 3.3.8.2 instead of LCO 3.3.8.1. These referential changes have been made.

STD DEP 16.3-24, LCO 3.10.3, Control Rod Withdrawal - Hot Shutdown Bases

LCO Bases provides a list of other Special Operations LCO applicable in MODE 3 with the reactor mode switch in the refuel position. In this list, “LCO 3.10.4, Control Rod Withdrawal-Cold Shutdown,” is listed. This Specification is applicable in MODE 4 with the reactor mode switch in the refuel position. Reference to the Specification is deleted since it does not apply.

STD DEP 16.3-25, LCO 3.9.1, Refueling Equipment Interlocks

LCO 3.9.1 requires the refueling equipment interlocks to be OPERABLE during in-vessel fuel movement with equipment associated with the interlocks. The refueling equipment interlocks (All-rods-in, Refuel machine position, and refuel machine main hoist, fuel loaded) are only applicable when the reactor mode switch is in the refuel position.

The LCO Background states, “With the reactor mode switch in the shutdown or refueling position, the indicated conditions are combined in logic circuits to determine if all restrictions on refueling equipment operations and control rod insertion are satisfied.”

This Background implies the instrumentation is applicable when the reactor mode switch is also in the Shutdown position. The changes provided in the LCO and Applicability both in the Specifications and Bases provides additional clarity on when the requirements are required to be met. This change is acceptable because the reactor mode switch Shutdown position requirements in LCO 3.3.5.1 ensures a control rod block is ensured.

STD DEP 16.3-26, LCO 3.10.2, Reactor Mode Switch Interlock Testing

The Bases Background for LCO 3.10.2 discusses the reactor mode switch positions and the related scram interlock functions. The discussion includes reference to a reactor high water level scram. The ABWR design does not include a “reactor high water level scram.” Therefore, Bases Section B 3.10.2 has been modified to remove reference to the scram for consistency with the DCD. Additionally, for each mode switch position, supplemental information regarding the neutron monitoring system scram and turbine control valve fast closure and turbine stop valve closure scram is added to clarify and enhance the Bases background discussion consistent with DCD Tier 2, Subsection 7.2.1.1.6.2.

STD DEP 16.3-27, LCO 3.10.2, Reactor Mode Switch Interlock Testing

Bases B 3.10.2 of Technical Specification (TS) LCO 3.10.2, Reactor Mode Switch Interlock Testing, provides a listing of other Special Operations LCOs applicable in MODES 3, 4, and 5. This listing includes LCO 3.10.7, Control Rod Testing - Operating. However, this Specification is applicable in MODES 1 and 2 with LCO 3.1.6 not met. Reference to the Specification is deleted since it does not apply. However, additional Specifications LCO 3.10.1, LCO 3.10.5, LCO 3.10.6, LCO 3.10.8, LCO 3.10.11, and LCO 3.10.12 have been added as references since they are applicable in MODES 3, 4, and 5.

STD DEP 16.3-28, LCO 3.10.1, In-Service Leak and Hydrostatic Testing Operation

The Applicable Safety Analyses description states, “The consequences of a steam leak under pressure testing conditions, with secondary containment OPERABLE, will be conservatively bounded by the consequences of the postulated main steam line break outside of secondary containment accident analysis described in Reference 2.” Reference 2 is DCD Tier 2, Section 15.1. The postulated main steam line break outside of secondary containment analysis is not discussed in Section 15.1. It is discussed in DCD Tier 2, Section 15.6.4. Therefore, the appropriate reference has been incorporated.

STD DEP 16.3-29, LCO 3.6.4.1, Secondary Containment

The drawdown time in SR 3.6.4.1.4 is $\leq$ 120 seconds. This time has been extended for “120 seconds” to “20 minutes” to be consistent with Tier 1 Table 2.14.4 Item 4.a and the analysis in DCD Section 15.6.5.5.1, Fission Product Releases.

STD DEP 16.3-30, LCO 3.6.4.1, Secondary Containment

The Applicability description in the Bases states, “In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the primary or secondary containment.”

The words “primary or” have been deleted from the end of the last sentence since the applicability is only when moving fuel assemblies in the secondary containment.

STD DEP 16.3-31, LCO 3.6.4.3, Standby Gas Treatment (SGT) System

The Bases Background section description states, “The moisture separator is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the influent air stream to the adsorber section of the filter train to less than 70% whenever SGT System is in operation (Ref. 2).” Reference 2 is DCD Tier 2, Section 6.2.3. Details of the design of the SGT System are described in detail in DCD Tier 2 Section 6.5.1. Therefore, the appropriate reference has been incorporated.

STD DEP 16.3-32, LCO 3.6.2.1, Suppression Pool Average Temperature

Condition D requires entry when suppression pool average temperature is > 43.3°C but < 48.9°C. Required Action D.1 requires the determination of suppression pool average temperature < 48.9°C. ACTION E requires entry when suppression pool average temperature > 48.9°C. Required Action E.1 requires the unit to be depressurize the reactor within 12 hours.

ACTION D has been revised to require the “determination” of suppression pool average temperature instead of a verification that the temperature is ≤ 48.9°C and in the same condition the plant is required to be in MODE 4 in 36 hours. In addition, Condition D has been changed to require the temperature to be monitored whenever temperature is > 43.3°C instead of the specified range.

ACTION E is revised by deleting the requirement to be in MODE 4 since the requirement has been incorporated in ACTION D.

This change is necessary since the plant should not be in an operating MODE with temperature > 43.3°C. This is consistent with LCO 3.6.2.1.c which states that the suppression pool temperature should be ≤ 43.3°C when THERMAL POWER is ≤ 1% RTP.

The range in ACTION D has been changed since it is prudent to monitor suppression pool temperature whenever temperature is above 43.3°C not just when within the temperature range.

The Bases ACTIONS have been changed accordingly.

STD DEP 16.3-33, LCO 3.6.2.1, Suppression Pool Average Temperature

The Bases LCO 3.6.2.1 parts a and b states that suppression pool temperature requirements when THERMAL POWER is < 1%. The LCO states that these limits apply when THERMAL POWER is > 1%RTP. The signs in the Bases have been modified consistent with the requirements in the LCO.

STD DEP 16.3-34, LCO 3.6.1.6, Wetwell-to-Drywell Vacuum Breakers

The LCO Bases states “All eight of the vacuum breakers must be OPERABLE for opening. All wetwell-to-drywell vacuum breakers, however, are required to be closed (except during testing or when the vacuum breakers are performing the intended design function).” The SR 3.6.1.6.1 Bases states, “Each vacuum breaker is verified closed (except when being tested in accordance with SR 3.6.1.6.2 or when performing its intended function) to ensure that this potential large bypass leakage path is not present.”

The allowance in the Bases that the wetwell-to-drywell vacuum breakers may be opened during testing has been deleted since it is not stated in the LCO or SRs. LCO 3.6.1.6 is applicable for MODES 1, 2 and 3. The surveillance requirement for vacuum breaker functional testing SR 3.6.1.6.2 has a frequency of 18 months which is based on the need to perform the surveillance during an outage. The vacuum breakers can only be manually operated and are only accessible during an outage, and therefore these statements in the Bases in the LCO and SR 3.6.1.6.1 discussions are not applicable.

STD DEP 16.3-35, LCO 3.9.6, Reactor Pressure Vessel (RPV) Water Level

Technical Specification (TS) Bases B 3.9.6, Reactor Pressure Vessel (RPV) Water Level (Refueling Operations) references NUREG - 0831, Supplement 6, Section 16.4.2. as a requirement for the minimum water level of 7.0 meters above the top of the RPV flange to ensure a decontamination factor (DF) of 100 used in the accident analysis for iodine activity in FSAR Section 15.7.4. However, as stated in the DCD/Tier 2, Section 15.7.4, Regulatory Guide 1.25 (March 23, 1972) is the appropriate reference for key analyses assumptions including the DF of 100 (99% Iodine filter). Accordingly, NUREG - 0831 is removed as a reference and the references renumbered.

The DCD states that 25% of 10 CFR 100 limits are met consistent with NUREG - 0800, Section 15.7.4. The acceptance criteria contained in the NUREG is, “25 percent or less of the 10 CFR Part 100 exposure guideline values.” Therefore, TS Bases B 3.9.6 background value is changed from ”less than” 25% to “less than or equal to” 25% of 10 CFR 100 limits.

In the Applicable Safety Analyses section of B 3.9.6 “dropped” fuel is changed to “damaged” fuel to more accurately describe the accident results and terminology of the analyses in DCD Section 15.7.4.

Finally, reference to LCO 3.7.6 (Fuel Pool Water Level) is made in the Applicability section of the Bases. This LCO numbering has been corrected to 3.7.8.

STD DEP 16.3-36, LCO 3.6.2.3, Residual Heat Removal (RHR) Suppression Pool Cooling

The Bases Background discussion states, “S/RV leakage, and high pressure core injection and Reactor Core Isolation Cooling System testing increase suppression pool temperature more slowly.” The ABWR design does not include a “high pressure core injection system” therefore it has been deleted from the Background discussion.

STD DEP 16.3-37, LCO 3.6.2.3, Residual Heat Removal (RHR) Suppression Pool Cooling

The Bases Reference 2 is the “ASME Boiler and Pressure Vessel Code, Section XI.” This Reference is not used in the discussion and it has been deleted.

STD DEP 16.3-39, LCO 3.3.4.2, Feedwater and Main Turbine Trip Instrumentation

This departure summary is relocated to Part 7, Section 2.2.1.

STD DEP 16.3-40, LCO 3.8. 2, AC Sources-Shutdown

The Bases LCO specifies the requirements for the requirements for OPERABILITY of the diesel generator. It states, “The Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as: DG in standby with the engine hot, DG in standby parallel test mode." The last sentence has been changed to "These capabilities are required to be met from a variety of initial conditions such as: DG in standby with the engine hot, with the engine at ambient conditions, or DG operating in standby parallel test mode." The change is made to be consistent with the Bases of LCO 3.8.1 and 3.8.11.

STD DEP 16.3-41, LCO 3.8.2, AC Sources-Shutdown

The Required Action for Condition A includes a Note that states, "Enter applicable Condition and Required Actions of LCO 3.8.10, with one required division de-energized as a result of Condition B." Condition B has been changed to Condition A since it is the intent of the Note as indicated in the associated Bases discussion. This change is a typographical misstatement in NOTE for CONDITION A in the REQUIRED ACTION section. The note erroneously requires entry into applicable Condition and Required Actions of LCO 3.8.10 with one required division de-energized as a result of Condition B. The entry should be as a result of Condition A.

STD DEP 16.3-42, LCO 3.8.4, DC Sources - Operating

The Bases of Required Action D.1 and D.2 states that "If all inoperable DC electrical power subsystems cannot be restored to OPERABLE status within the associated Completion Times for Required Action A.1, B.2, and C.1 or C.2, the unit must be brought to a MODE in which the LCO does not apply." This sentence has been changed to "If all inoperable DC electrical power subsystems cannot be restored to OPERABLE status within the associated Completion Times, the unit must be brought to a MODE in which the LCO does not apply." This change is made to be consistent with the Required Action.

STD DEP 16.3-43, LCO 3.6.1.1, Primary Containment

The Bases Background states "The primary containment air lock is OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Locks." The ABWR Containment has two airlocks. Therefore, "air lock is" is changed to "air locks are." This change is made to be consistent with other LCOs and the containment design.

STD DEP 16.3-44, LCO 3.6.1.1, Primary Containment

The Bases states, "Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions. Failure to meet air lock leakage testing (SR 3.6.1.2.1), [resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.7),] or main steam isolation valve leakage (SR 3.6.1.3.13), or hydrostatically tested valve leakage (SR 3.6.1.3.12) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of 10 CFR 50, Appendix J."

The main steam isolation valve leakage SR has been eliminated from this list since the containment analyses assumes a specific leakage limit for L_a and a specific leakage limit for main steam isolation valve leakage. Therefore, main steam line leakage is excluded from the L_a term.

STD DEP 16.3-45, LCO 3.6.1.1, Primary Containment

The Background Section states “This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of References 1 and 2.”

Reference 1 is Tier 2 Section 6.2 “Containment Systems” and Reference 2 is Tier 2 Section 15.1. “Decrease in Reactor Coolant Temperatures.” Section 15.1 is not the appropriate reference and it has been deleted and replaced with DCD Tier 2, Section 15.6 “Decrease in Reactor Coolant Inventory.”

STD DEP 16.3-46, LCO 3.7.2, RCW, RSW, and UHS Applicability

The Applicability of 3.7.2 is in MODE 5 except with the reactor cavity to dryer/separator storage pool gate removed and water level ≥ 7.0 m over the top of the reactor pressure vessel flange. The Applicability of LCO 3.7.3 is MODE 5 with the reactor cavity to dryer/separator storage pool gate removed and water level ≥ 7.0 m over the top of the reactor pressure vessel flange. The Applicability requirements of these supporting system Specifications should match the Applicability of the supported system Specifications 3.9.7 and 3.9.8. The applicability of Specification 3.9.7 is MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level ≥ 7.0 m above the top of the RPV flange while the Applicability of LCO 3.9.8 is MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level < 7.0 m above the top of the RPV flange. The Applicability of Specification 3.7.2 has been changed to be consistent with the Applicability of 3.9.8 while the Applicability of Specification 3.7.3 has been changed to be consistent with the Applicability of 3.9.7. The Bases Background, Applicable Safety Analyses, LCO, and Applicability have also been corrected. In addition, the Bases discussion for SR 3.7.3.5 of the DCD states that the Logic System Functional Test (LSFT) in SR 3.3.5.1.4 overlaps this SR, which is incorrect. The correct references to the SRs in LCO 3.3.1.1 and LCO 3.3.1.4 are incorporated.

STD DEP 16.3-47, LCO 3.7.4, Control Room Habitability Area (CRHA)-Emergency Filtration (EF) System

SR 3.7.4.4 requires verification that each EF division can maintain a positive pressure of > 3.2 mm water gauge relative to the atmosphere during the isolation mode of operation at a flow rate of < 360 m^3/h. The flow rate of 360 m^3/h has been changed to 3400 m^3/h to be consistent with Tier 1 Table 2.15.5a Item 5.b. A similar change is made to the associated Bases of SR 3.7.4.4.

STD DEP 16.3-48, LCO 3.7.4, Control Room Habitability Area (CRHA)-Emergency Filtration (EF) System

The Bases Background states, “Each division consists of an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, two 100% capacity fans, and the associated ductwork and dampers.” The Bases has been revised to indicate that the CRHA System includes two

100% capacity fans. Therefore the Bases has been modified to reflect the actual design.

STD DEP 16.3-49, LCO 3.8.1, AC-Sources-Operating

Note (b) of Table 3.8.1-1, Diesel Generator Test Schedule states that maintaining the table-specified DG test interval “until seven failure-free starts from standby conditions and load and run tests have been performed” is consistent with Regulatory Position of Regulatory Guide 1.9, Revision 3. The referenced Regulatory Guide has no Regulatory Position that specifies the seven consecutive failure-free starts credited to the Regulatory Position. The mention of the seven consecutive failure-free starts does appear in Generic Letter 84-15, “Example Technical Specifications” regarding DG testing. DG testing is not effected by this reference change. The change only deletes the statement that the test interval is consistent with the Regulatory Position in R.G. 1.9.

STD DEP 16.3-50, LCO 3.3.1.4, ESF Actuation Instrumentation

The Applicable Modes or Other Specified Conditions for Table 3.3.1.4-1, Function 14.a and 14.b, is changed to include MODE 1. This is consistent with the Applicable Modes or Other Specified Conditions for Table 3.3.1.1-1, Function 3.b (i.e., MODE 1 also).

STD DEP 16.3-51, LCO 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air

The Bases Background states “Each DG has an air start system with adequate capacity for five successive start attempts on the DG without recharging the air start receiver(s).” The actual design will include two redundant DG air start subsystems, each with adequate capacity for five successive start attempts on the DG without recharging the air start receiver(s). LCO 3.8.3 ACTION E, the BASES Background and the BASES LCO have been modified to reflect this clarification.

STD DEP 16.3-52, LCO 3.8.8, Inverters - Shutdown

The Applicable Safety Analyses states, “The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protection System (RPS) and Emergency Core Cooling Systems (ECCS) instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.”

The statement is revised to indicate that the inverters supply the Class 1E CVCF loads. This change is consistent with the description in the Applicable Safety Analyses of LCO 3.8.7, Inverters-Operating.

STD DEP 16.3-55, LCO 3.3.4.1, Anticipated Transient Without Scram (ATWS) and End-of-Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

The Surveillance Requirement discussion has been changed to revise the time of EOC-RPT System Response Time to RPT System Response Time. The title and definition is being revised since ATWS and EOC response times are credited in the transient and accident analyses.

STD DEP 16.3-57, LCO 3.3.1.2, Reactor Protection System (RPS) and Main Steam Isolation Valve (MSIV) Actuation

The Bases discussion for Required Action I.1 and I.2 is changed to remove the statement, “Note that the automatic actuation logic becomes 1/3 in this condition so there is an increased vulnerability to spurious trips” because it is incorrect. The automatic actuation logic is unaffected by placing the affected division in trip per Action I.1.

STD DEP 16.3-58, LCO 3.8.6, Battery Cell Parameters

Condition A requires entry when “One or more batteries with one or more battery cell parameters not within limits.” Condition A has been changed to require entry when “One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category A or B limits.”

Table 3.8.6-1 includes Category A, B, and C limits. Condition B, in part, requires entry when “One or more batteries with one or more battery cell parameters not within Category C limits.” The change which adds “Table 3.8.6-1 Category A or B” to CONDITION A is a clarification that the limits within which the battery cell parameters must be maintained.

STD DEP 16.3-59, LCO 3.3.6.2, Remote Shutdown System

Function 13 of Table 3.3.6.2-1 is the “RPV Narrow Range Water Level.” It has been changed to “RPV Shutdown Range Water Level.” This change is consistent with DCD Section 7.4.1.4.4. Similar changes have been made to the Bases.

STD DEP 16.3-60, LCO 3.3.6.2, Remote Shutdown System

RSW Strainer Differential Pressure Instrumentation has been added to the list of Remote Shutdown System parameters monitored as Function 17 consistent with DCD Figure 7.4-2.

STD DEP 16.3-61, LCO 3.3.7.1, CRHA EF System Instrumentation

Table 3.3.7.1-1 includes two Footnotes that are not referenced in the Table. Footnotes (a) and (b) to Table 3.3.7.1-1 have been deleted. The Footnotes are associated with the Applicability of the instrumentation. Since the Applicability of the instrumentation is covered in the Applicability statement the Footnotes are not needed.

STD DEP 16.3-62, LCO 3.3.8.1, Electric Power Monitoring

ACTION C requires entry when “Required Action and associated Completion Time of Condition A or B is not met in MODE 1, 2, or 3."

ACTION D requires entry when “Required Action and associated Completion Time of Condition A or B is not met in MODE 4 or 5.”

The associated Bases refer only to Condition B. Therefore, Condition A has been added to the Bases descriptions for ACTIONS C and D.

STD DEP 16.3-63, LCO 3.3.8.2, Reactor Coolant Temperature Monitoring-Shutdown

The Bases Background states, “The temperature monitoring instrumentation will provide temperature indication and trends to the operator in the main control room during RHR decay heat removal operation. One temperature monitoring for each RHR channel is available to monitor reactor coolant temperature at the inlet to the RHR heat exchanger.” The word “transmitter” has been added after monitoring to be consistent with the terminology being used.

STD DEP 16.3-64, LCO 3.3.5.1, Control Rod Block Instrumentation

Required Action B.2 states, “Verify RCIS blocks control rod movement by attempting to withdraw one rod or one gang or rods.” The sentence is changed to “Verify RCIS blocks control rod movement by attempting to withdraw one rod or one gang of rods.” Changes typographical misstatement in Required Action B.2 from “or” to “of” in referring to the withdrawal of one gang “of” control rods.

STD DEP 16.3-65, LCO 3.3.5.1, Control Rod Block Instrumentation

Changes typographical misstatement in the NOTE for SR 3.3.5.1.1 wherein the CHANNEL FUNCTIONAL TEST is not required to be performed until 1 hour after THERMAL POWER is >30% RTP. The 30% value, rather than the 10% value, in the NOTE is in agreement with TS Table 3.3.5.1-1, note (a) for the RCIS Automated Thermal Limit Monitor function.

STD DEP 16.3-66, LCO 3.3.5.1, Control Rod Block Instrumentation

Changed the number of reactor mode switch position channels required to be OPERABLE when the reactor mode switch is in the shutdown position from three to four channels. This change is in agreement with TS Table 3.3.5.1-1, Function 2, Reactor Mode Switch - Shutdown Position requirements for required channels.

STD DEP 16.3-67, LCO 3.3.5.1, Control Rod Block Instrumentation

Changes typographical misstatement in ACTION E.1 and E.2 by removing the word “in” in the phrase, “If there are failures 'in' of the Reactor Mode Switch - Shutdown Position Function the plant must be placed in a condition where the LCO does not apply.” Also, corrects the grammatical misstatement in the next sentence where, “…and initiating 'to fully inserting' of all…” should be changed to read, “…and initiating full insertion of all…”.

These changes do not change the meaning or intent of these statements.

STD DEP 16.3-68, LCO 3.1.3, Control Rod OPERABILITY

The Bases description of Required Actions A.1, A.2, and A.3 states, “If the motor is working and the rod is actually stuck, the traveling nut will back down from the bottom of the drive and a rod separation alarm and rod block will result (see LCO 3.3.5.1).” Reference to Specification 3.3.5.1 has been deleted. The rod separation alarm and rod block are not included in LCO 3.3.5.1, Control Rod Block Instrumentation. However, LCO 3.3.5.1 does include other Control Rod Block Functions.

STD DEP 16.3-69, LCO 3.6.1.2, Primary Containment Air Locks

Required Action B.2 states, “Lock an OPERABLE door closed in the affected air lock( ).” The ABWR Containment has two airlocks. Therefore, “air lock( )” is changed to “air lock(s).”

STD DEP 16.3-70, LCO 3.6.1.2, Primary Containment Air Locks

The Bases Background discussion states, “SR 3.6.1.1.1 leakage rate requirements conform with 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions.” Specification 3.6.1.2 is associated with primary containment air locks. Therefore, the appropriate SR to be utilized in the discussion is SR 3.6.1.2.1 not SR 3.6.1.1.1.

STD DEP 16.3-71, LCO 3.6.1.3, Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 includes two Surveillance concerning the status of the containment purge valves. SR 3.6.1.3.1 requires the primary containment purge valves to be “closed and sealed.” SR 3.6.1.3.2 requires the same valves to be “closed” however a Note allows the valves to be opened when the valves are being used for inerting, de-inerting, pressure control, ALARA, or air quality considerations for personnel entry, or Surveillances that require the valves to be open.

Utilizing the Note in SR 3.6.1.3.2 would always be a failure to meet SR 3.6.1.3.1. The ABWR utilizes an inerted containment and therefore, SR 3.6.1.3.2 is the appropriate SR for the design.

This change results in a number of changes:

- SR 3.6.1.3.1 is deleted and subsequent SRs have been renumbered;
- 3.6.1.3 ACTION D has been replaced with a new Condition to cover Purge valve leakage rate, main steam isolation valve leakage, or hydrostatically tested line leakage not within limit. The Completion Time for the Condition has been bracketed until some operating experience is reviewed to determine whether the Completion Times are appropriate;
- Conditions A, B, F, G, and H have been revised to cover changes to Condition D.
- The Actions Note 1 has been modified to allow purge valve penetrations to be unisolated intermittently under administrative controls;
- Changes have been made to the Bases descriptions in the LCO, Applicability, ACTIONS, and Surveillances.

STD DEP 16.3-72, LCO 3.6.1.3, Primary Containment Isolation Valves (PCIVs)

ABWR DCD SR 3.6.1.3.13 specifies leakage rate limits for the main steam isolation valves. This SR contains a Note that states the results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. This Note has been deleted since the Containment

Radiological Analysis takes into account MSIV leakage separately from La. Corresponding changes are made to the SR 3.6.1.3.13 Bases.

STD DEP 16.3-73, LCO 3.6.1.3, Primary Containment Isolation Valves (PCIVs)

The Background states, “Two additional redundant excess flow isolating dampers are provided on the vent line upstream of the Standby Gas Treatment (SGT) System filter trains. These isolation dampers, together with the PCIVs, will prevent high pressure from reaching the SGT System filter trains in the unlikely event of a loss of coolant accident (LOCA) during venting. Closure of the excess flow isolation dampers will not prevent the SGT System from performing its design function (that is, to maintain a negative pressure in the secondary containment). To ensure that a vent path is available, a 50 mm bypass line is provided around the dampers.”

The statement has been corrected to reflect the DCD Tier 2 Figure 6.2-39, for the Atmosphere Control System. It now reads, “The PCIVs will close before fuel failure and prevent high pressure from reaching the SBT system filter trains in the unlikely event of a loss of coolant accident during venting.”

The Applicable Safety Analyses specifies assumptions used for the purge valves in the analyses. The changes to the description provide the appropriate assumptions for the analyses. The Applicable Safety Analyses also discusses the assumptions used for closure times in the radiological analyses. This statement has been deleted since the analysis assumes a leakage of La from the start of the accident.

STD DEP 16.3-74, LCO 3.6.1.3, Primary Containment Isolation Valves (PCIVs)

This departure corrects and clarifies the LCO 3.6.1.3 Bases discussion regarding testing, required actions, and surveillance requirements. The Bases states, “Purge valves with resilient seals, secondary bypass valves, MSIVs, and hydrostatically tested valves must meet additional leakage rate requirements.” The additional leakage rate requirements for the purge valves with resilient seals, MSIVs, and hydrostatically tested valves are provided in SR 3.6.1.3.7, SR 3.6.1.3.13, and SR 3.6.1.3.12, respectively. The words “secondary bypass valves” have been deleted from the phrase since there are no additional requirements for secondary bypass valve leakage.

Required Action A.2 Completion Time states that the verification of the isolation of the affected part must be verified “Once per 31 days for isolation devices outside primary containment, drywell, and steam tunnel. The Bases description in the Bases did not reflect the “drywell and steam tunnel.” Therefore the words have been added to the Required Action discussion.

ABWR DCD SR 3.6.1.3.9 (STP SR 3.6.1.3.8) states that the “LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.3.6" overlaps this SR to provide complete testing of the safety function.” This statement has been changed to “The testing in LCO 3.3.1.1 and LCO 3.3.1.4 overlaps this SR to provide complete testing of the safety function.” This change is appropriate since SR 3.3.6.3.6 does not exist.

STD DEP 16.3-75, LCO 3.7.6, Main Condenser Offgas

The Background section states “The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.” The description has been changed to “The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the charcoal adsorber vault.” This change is made since there is no hold-up line in the description of the offgas system in DCD Section 11.3.4 or identified in Offgas System Figure 11.3-2.

STD DEP 16.3-76, LCO 3.7.5, Control Room Habitability Area (CRHA) - Air Conditioning (AC) System

SR 3.7.5.2 requires the performance of an actual or simulated initiation test. The Bases do not include the associated discussion. The Bases have been modified accordingly.

STD DEP 16.3-77, LCO 3.3.6.1, Post Accident Monitoring (PAM) Instrumentation

The description of the Suppression Pool level instrumentation for Post Accident Monitoring in the technical specification bases has been changed to match the DCD ABWR certified design description.

STD DEP 16.3-78, LCO 3.3.6.1, Post Accident Monitoring (PAM) Instrumentation

The containment water level parameter has been removed from Post Accident Monitor technical specifications. The instrumentation does not meet the Bases' criteria for inclusion (i.e., Drywell water level is classified as Cat. 2 and sump level is classified as Cat 3). Also, the Bases only require that Post-Accident Monitoring instruments that are classified as Regulatory Guide 1.97 Type A or Category I be included. Lower drywell level instrumentation is described as “not warranted” in the DCD.

STD DEP 16.3-80, LCO 3.8.1, AC-Sources-Operating

The Bases Header for Required Actions D.1 and D.2 is not correctly located. The header was properly located to separate the Required Actions discussion for Required Actions C.4, C.5, and C.6 and Required Action D.1 and D.2.

STD DEP 16.3-81, LCO 3.3.1.2, Reactor Protection System (RPS) and Main Steam Isolation Valve (MSIV) Actuation

The Applicable Modes or Other Specified Conditions for SRNM and APRM LOGIC CHANNELS (Function 1a) has been added as footnote (b) to Table 3.3.1.2-1. The footnote appropriately requires the SRNM and APRM LOGIC CHANNELS to be OPERABLE when the associated Function in LCO 3.3.1.1 is required to be OPERABLE.

The above can be clarified further. In Table 3.3.1.1-1, note that Functions 1a, 1b, and 1d sensor instrumentation are required to be Operable in MODE 5. The other subfunctions (1c, 2a, 2b, 2c, 2d, and 2e) are not required to be Operable in MODE 5. Therefore, footnote (b) is needed to clarify that RPS Actuation Logic Channels are only required to be Operable in MODE 5 for Functions 1a, 1b, and 1d in Table 3.3.1.1-1.

These are the NMS “associated” functions in LCO 3.3.1.1 that are required to be operable in MODE 5.

Divisional Functional Test, SR 3.3.1.2.2, was incorrectly applied to Output Channels (in Items 1 and 2 of Table 3.3.1.2-1, RPS Actuation, and MSIVs and MSL Drain Valves Actuation) in the DCD and has been deleted. The Definition of Divisional Functional Test in the original DCD was: “The injection of simulated or actual signals into a division as close to the sensors as practicable to verify Operability of Sensor Channels and Logic Channels in that division. The Divisional Functional Test may be performed by means of a series of sequential or overlapping steps. The test shall comprise all of the equipment from the DTM (now changed to DTF) inputs to Logic Channel outputs….". From this definition, SR 3.3.1.2.2 should not be applied to Output Channels.

The bases discussion in the APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY has also changed based upon the changes to the specification.

STD DEP 16.3-82, LCO 3.3.1.2, Reactor Protection System (RPS) and Main Steam Isolation Valve (MSIV) Actuation

This departure is a clarification to Technical Specification (TS) B 3.3.1.2/LCO 3.3.1.2, Actions for Conditions B, F, J, K and L. The change is to specify and clarify that 1) Condition B or F occurs if two logic channels or two output channels “for the same Function” become inoperable, 2) Conditions J and K assure that appropriate actions are taken for “one or more” inoperable RPS Actuation Functions, and 3) Condition L assures appropriate actions are taken for “one or more” inoperable MSIV Actuation Functions. (Note: These changes are required in order to be consistent with the Conditions description for LCO 3.3.1.2 as shown in the Technical Specifications in COLA Part 2, Tier 2, Subsection 16.3.3.1.2.)

STD DEP 16.3-83, LCO 3.3.1.3, Standby Liquid Control (SLC) and Feedwater Runback (FWRB) Actuation

The bases discussion of the Manual ATWS-ARI/SLCS Initiation logic has been changed to describe the actual plant design. This Manual ATWS-ARI/SLCS discussion is illustrated in DCD Figures 15E-1a and 15E-1b. The Manual ATWS-ARI/SLCS Initiation originates at the Manual ATWS A and Manual ATWS B pushbuttons shown on DCD Figure 15E-1a. Each pushbutton represents a manual initiation channel with input from both switches required to satisfy the manual actuation logic.

STD DEP 16.3-84, LCO 3.3.1.1, SSLC Sensor Instrumentation

The Applicable Conditions for Function 3c, Reactor Vessel Steam Dome Pressure - High, and Function 7c, SLCS and FWRB Initiation, in Table 3.3.1.1-1, are changed from Condition G to Condition H. The Applicable Modes or Other Specified Conditions for Functions 3c and 7c are MODE 1 and MODE 2, and the change to Condition H ensures that the Applicable Modes or Other Specified Conditions for these Functions are exited in the event that Condition E is entered. That is, per Condition E, if the Required Action and associated Completion Time of Condition A, B, C, or D are not met, the appropriate Required Action in Table 3.3.1.1-1 for the Function is H (i.e., Be in Mode 3).

STD DEP 16.3-85, LCO 3.3.1.1, SSLC Sensor Instrumentation

The Bases discussion for the Automatic Depressurization System (ADS) is changed to correct the ADS accumulators' capacity to operate the safety relief valves with no external source of nitrogen.

The text change, supported by DCD Sections 7.3.1.1.1.2(3)(paragraph 2) and 5.2.2.4.1, states that the ADS accumulators have sufficient capacity to operate the safety relief one time at drywell design pressure or five times at normal drywell pressure with no external source of nitrogen.

STD DEP 16.3-86, LCO 3.3.1.4, ESF Actuation Instrumentation

The text of SR 3.3.1.4.7 is changed to apply the surveillance to both the manual initiation and manual inhibit channels for ADS. Table 3.3.1.4-1, Functions 4.c and 4.f both reference this surveillance requirement. Footnote (d) Table 3.3.1.4-1 is also changed to reflect both manual initiation and manual inhibit channel Functions.

STD DEP 16.3-87, LCO 3.3.1.4, ESF Actuation Instrumentation

The Bases discussion for Required Action G.1 is changed to correct the Conditions, that if not met within the specified Completion Times, result in entry into Condition and performance of Required Action G.1. The text change will make the Specification and its Bases agree. The correct Conditions, B, C, D, E, or F, are specified in TS Bases 3.3.1.4 Condition G.

STD DEP 16.3-89, LCO 3.1.2, Reactivity Anomalies

Reference to the rod drop accident has been deleted from the Applicable Safety Analyses of TS Bases 3.1.2, Reactivity Anomalies. The event is not postulated to occur for the ABWR. This is consistent in the discussion in the ABWR DCD Section 15.4.10.3.1. This Section states, in part, there is no basis for the control rod drop event to occur.

STD DEP 16.3-90, LCO 3.1.3, Control Rod OPERABILITY

The Applicable Safety Analyses states that the analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, 4, and 5. Reference 5 is the rod ejection accident. In accordance with Reference 5 the event is not postulated to occur, therefore it has been deleted.

STD DEP 16.3-91, LCO 3.3.1.1, SSLC Sensor Instrumentation

The incorrect title of Function 33, “Control Building Basement Equipment Cubicle”, in the Bases discussion is changed to its correct title, “RCW/RSW Heat Exchanger Room Water Level - High”.

STD DEP 16.3-92, LCO 3.3.1.1, SSLC Sensor Instrumentation

The Bases discussion for Required Actions P.1, P.2, R.1, and R.2 is changed to include other conditions (e.g., not placed in trip, not isolated) that also result in entering the specified actions.

STD DEP 16.3-93, LCO 3.3.1.1, SSLC Sensor Instrumentation

A typographical misstatement in the Bases discussion for SR 3.3.1.1.10 and 3.3.1.1.11 is changed to correct the specified SR number. Changed SR 3.2.1.1.10 to read SR 3.3.1.1.10.

STD DEP 16.3-94, LCO 3.3.1.4, ESF Actuation Instrumentation

The Applicable Modes or Other Specified Conditions for Function 13c, CUW Isolation and SLC Initiation, in Table 3.3.1.4-1 is changed from MODE 1, MODE 2, and MODE 3, to only MODE 1 and MODE 2, since these are the MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC system in LCO 3.1.7.

STD DEP 16.3-95, LCO 3.2.3 Linear Heat Generation Rate (LHGR) (Non-GE Fuel)

STP 3&4 Technical Specification LCO 3.2.3, Linear Heat Generation Rate (LHGR) (Non-GE Fuel) and its Bases are deleted from COLA Part 2, Tier 2, Chapter 16 and Part 4 because DC/COL-ISG-08, “Interim Staff Guidance - Necessary Content of Plant-Specific Technical Specifications When a Combined License Is Issued” requires the removal of all brackets within the Technical Specifications prior to granting a COL, and this Specification cannot stand on its own without the use of brackets if GE fuel is specified. As presently written, the COL application calls for GE fuel. Non-GE Fuel is not presently applicable and therefore, this Specification is not currently needed. It is anticipated that a license amendment request will be submitted to reinstate this Specification if a non-GE fuel type is chosen in the future.

STD DEP 16.3-96, LCO 3.4.1, RIPs Operating

The optional flexibility to operate with fewer than 9 reactor internal pumps (RIPs) is not currently supported by analysis; therefore, the bracketed optional flexibility in LCO 3.4.1 is being removed. The Technical Specification LCO 3.4.1 information being removed is all contained within brackets; and therefore, its removal does not require an exemption per Section VIII.C.4 of Appendix A to 10 CFR Part 52. However, the Bases for this Specification contain several paragraphs and/or sentences describing this optional flexibility that are not contained within brackets. DC/COL-ISG-08, “Interim Staff Guidance - Necessary Content of Plant-Specific Technical Specifications when a Combined License is Issued,” requires the removal of all brackets within the Technical Specifications prior to granting a COL. This departure is written to remove the Technical Specification 3.4.1 Bases information related to this optional flexibility (paragraphs 2 and 3 under “Applicable Safety Analyses,” the second sentence of the single paragraph under “LCO” and Reference 3, requiring a plant specific analysis for fewer than 9 RIPs operating.