3.0 Departures Not Requiring Prior NRC Approval

The departures from Tier 2 information summarized in this section of COLA Part 7 do not involve a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications. The departures:

- do not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated
- do not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated
- do not result in more than a minimal increase in the consequences of an accident previously evaluated
- do not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated
- do not create a possibility for an accident of a different type than any evaluated previously
- do not create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously
- do not result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered
- do not result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses
- do not result in a substantial increase in the probability of an ex-vessel severe accident such that a particular ex-vessel severe accident previously reviewed and determined to be not credible could become credible
- do not result in a substantial increase in the consequences to the public of a particular ex-vessel severe accident previously reviewed

Therefore, in accordance with Regulatory Guide 1.206, Section C.IV.3.3 and with 10 CFR 52, Appendix A, Section VIII.B.5, these departures do not require prior NRC approval or an exemption from 10 CFR 52, Appendix A.

STD DEP 1.1-1, Type of License Required

Description

The reference ABWR DCD was submitted to receive a design certification. The COL applicant submits a site-specific DCD to receive a Class 103 combined operating license under 10 CFR 52.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This change only updates the DCD to reflect the type of license for which the applicant is applying. There is no change in any design or function of an SSC important to safety as described in the DCD as a result of this change. Consequently, this change has no impact on the frequency or consequences of any accident or malfunction of an SSC important to safety previously evaluated. There is no impact on the frequency or consequences of any ex-vessel severe accident previously reviewed. This change has no impact on any Tier 1, Tier 2*, Technical Specifications, bases for the Technical Specifications, or operational requirements information.

As a result of this evaluation, prior NRC approval of this change is not required.

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

Description

The reference ABWR DCD is based on a single-unit site. Because STP 3 & 4 is a dual-unit project on an existing site, some supporting systems described in the DCD are single systems that support two or more units. In addition, STP 3 & 4 share the main cooling reservoir with STP 1 & 2.

The systems shared by STP 3 & 4 include:

- Fire Protection Water Supply System - Regulatory Guide 1.189, Rev. 1, allows for use of a common water supply at multi-unit nuclear power plant sites. A single fire protection pump house and two storage tanks provide water for fire suppression to both units via piping in the yard. Since STP 3 & 4 do not share fire areas where safe shutdown systems are located, and it is extremely unlikely that there will be simultaneous fires in areas of the plant affecting safe shutdown areas, it is extremely unlikely that protection systems for both units will need to function at the same time.

- A common nonsafety-related communication system is required for multi-unit sites to provide plant wide communications. A common communication system providing plant wide communications is a personnel safety enhancement since it allows for ease of communication between units.

- Makeup Water Preparation - A common nonsafety-related makeup water preparation system that utilizes a common raw water storage tank and a common demineralized water storage tank will supply water to the makeup water condensate system and makeup water purified system of both units. This system is discussed further in STP DEP 9.2-2. Sharing of the MWP System does not impair the ability to cooldown STP Units 3 & 4 under Station Blackout conditions. The Station Blackout analysis is contained in Appendix 1C of the ABWR DCD. The primary source of water during the initial 10 to 60 minute period of a Station Blackout event is from the Condensate Storage Tank (CST) for each unit. The source of water for each CST is the shared MWP System via the Makeup Water Condensate (MUWC) System. During a Station Blackout, each unit's respective CST is capable of providing at least 8 hours of makeup water without replenishment. The Alternate AC power source (i.e., the CombustionTurbine Generator) for each unit is designed to start and load 10 minutes into the event. With the use of Alternate AC power sources other water sources (including MWP) are readily available for makeup, heat removal, and plant equipment cooling.

- Hydrogen Gas Storage Facility - A single nonsafety-related bulk hydrogen gas storage facility will be used to store hydrogen compressed gas cylinders for two units. The bulk hydrogen storage facility will be located at least 100m from any safety-related building or structure to prevent damage to safety-related equipment due to a fire or explosion at the facility.

- A common plant grounding grid is used that extends the contact area to ground and meets the resistance-to-ground criterion. The system in electrically interconnected between units.

- Potable Water system is shared between STP 3 and 4 and the Sanitary Treatment system are shared between all four units on site as well as with common buildings. This is discussed further in STP DEP 9.2-8.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change.

The functional description of each of the systems proposed to be shared between STP 3 & 4 that are affected by this change (Fire Protection System Water Supply, Communications System, Makeup Water Preparation, Potable and Sanitary Water, Bulk Hydrogen Gas Storage Facility, and common grounding grid) is not significantly changed by this departure. Each system is consistent with the description contained in the ABWR DCD except that each will be sized and designed to serve two units instead of one standard ABWR.

The proposed changes to these common systems to both Units 3 & 4 do not involve a reduction in their ability to support the mitigation of an accident or a malfunction of equipment important to safety in that they will not impede required actions by Engineered Features designed for this purpose. In the case of the shared Fire Water

Supply System, the occurrence of simultaneous fires in separate fire areas containing safe shutdown equipment in either or both units simultaneously is extremely unlikely.

Changes associated with this departure do not affect fission product barriers. These changes do not affect the probability of occurrence of a severe accident as described by the DCD, nor do they increase the consequences of a severe accident.

Additionally, the Ultimate Heat Sink that is specific to each unit (i.e., not shared) is consistent with the approved ABWR DCD.

Based on this evaluation, prior NRC approval of the change is not required.

STD DEP 1.2-1, Control Building Annex

Description

The Reactor Internal Pump (RIP) motor-generator sets and associated support components are relocated to a new, Non-Seismic Category I Control Building Annex adjacent to the Control Building. There was insufficient space in the CB for the two RIP MG sets and their associated equipment. This departure creates no new adverse effects and eliminates potential adverse effects that were identified for the standard design.

Evaluation Summary

The Control Building Annex is a nonsafety-related structure located adjacent to the Control Building. It houses the two reactor internal pump motor-generator sets, control panels, and the cooling water lines, HVAC system, and electrical lines that support the motor-generator sets. As described in DCD Tier 2 Section 9.5.10.3, the reactor internal pump motor-generator set equipment performs no safety-related function. Failure of the motor-generator set equipment does not compromise any safety-related system or component and does not prevent safe reactor shutdown.

The Control Building Annex has no personnel or equipment access paths to the Reactor Building. The Control Building Annex has one access path which is used for both personnel and equipment. This access path has a watertight door installed at the entrance to the Control Building designed to protect the Control Building from flood effects external to the Control Building. This includes protection from the effects of internal flood initiated within the Control Building Annex or the effect of external floods due to natural phenomena. Any penetrations between the Control Building and the Control Building Annex are either above any design basis flood levels or are designed to be watertight. This approach is consistent with FSAR Tier 2 Section 3.4.3.1 that addresses penetrations and doors that penetrate the exterior walls of Seismic Category I (safety-related) buildings. Therefore, flooding in the Control Building Annex won't have any effect on any safety-related buildings.

The CB Annex is also a Non-Seismic Category I building, but is designed to withstand the SSE to avoid jeopardizing adjacent Seismic Category I buildings.

Consequently , there is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety. Furthermore, there is no impact on fission product barriers or the probability of an ex-vessel severe accident. Therefore, this change has no adverse impacts and does not require prior NRC approval.

This change meets the criteria of 10 CFR 52 VIII.B.5. This change does not affect Tier 1, Tier 2*, or Technical Specifications or operational requirements. Therefore prior NRC approval of this change is not required.

STP DEP 1.2-2, Turbine Building

Description

The Turbine Building design has changed because of the following:

- The turbine generator described in the reference ABWR DCD is now obsolete and the replacement will differ dimensionally. The turbine cycle equipment such as feedwater heaters and pumps also differ from the cycle equipment described in the DCD.
- The power generation heat sink described in the DCD (natural draft cooling tower) is being replaced by a cooling reservoir. This affects the sizing of the condenser and circulating water piping. The design now includes condensate booster pumps.
- The DCD medium voltage electrical system design is being replaced by a dual voltage design and requires relocation of major components into and within the Turbine Building.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The change affects the function, but is bounded by the safety analysis.

The Turbine Building (T/B) s a nonsafety-related structure located adjacent to the Control Building. The T/B includes the electrical building and houses the main turbine generator and other power conversion cycle equipment and auxiliaries. With the exception of instrumentation associated with Reactor Protection System (RPS) and the safety-related condensate pump motor trip circuit breakers, there are no safety-related equipment in the T/B. The electrical building houses various plant support systems and equipment such as non-divisional switchgear and chillers.

Since the safety-related condensate pump motor trip circuit breakers are located above the design basis flood level and the T/B is designed to withstand the SSE, this change does not result in any increase in the frequency of an accident previously evaluated.

The Circulating Water System (CWS) and the Turbine Building Service Water System (TSW) are the only systems large enough to fill the condenser pit; therefore, only these two systems are required to be addressed to show that the T/B design is adequate to prevent flooding into the adjacent Reactor Building and Control Building. The CWS and TSW floods are limited by system isolation signals from leak detectors in the condenser pit and the TSW System equipment room. The increased area in T/B provides adequate volume for storing the limited flooding water from CWS and TSW to assure that water level remains below the access level to R/B and C/B via Service Building. Therefore, this change has no adverse impact to the result of DCD T/B flooding protection analysis evaluation, which means the flooding won't have any effect on any safety-related buildings.

The T/B is also a Non-Seismic Category I building, but is designed to withstand the SSE to avoid jeopardizing adjacent Seismic Category I buildings.

Consequently, this change does not result in any increase in the frequency of a malfunction of a structure, system, or component (SSC) important to safety.

This change affects the function, but is bounded by the safety analysis and as discussed above has no adverse impact. It does not require prior NRC approval.

STD DEP 1AA-1, Shielding Design Review

Description

Appendix 1AA of the reference ABWR DCD provides the integrated doses for environmental qualification of safety-related equipment. These doses have been re-evaluated in the STP 3 & 4 FSAR using similar regulatory guidance, but incorporating the results of design detailing. The doses for the ECCS pump rooms and the SGTS area increase compared to the original DCD values. Safety-related equipment will be qualified to the increased values as required.

Evaluation Summary

This departure is the result of re-evaluation of the post-accident radiation conditions inside the reactor building. There is no impact on the frequency of occurrence or the consequences of an accident as a result of this change. In addition, any SSCs important to safety will be qualified to the revised radiation dose limit, so there is no impact on the likelihood of occurrence or consequences of a malfunction of an SSC important to safety as a result of this change.

There is no new accident scenario or no unexpected malfunction of an SSC important to safety as a result of this change. A design basis limit for a fission product barrier is not exceeded or altered. A method of evaluation in establishing the design bases or in the safety analyses does not change. There is no impact on the probability or the consequences to the public of an ex-vessel severe acident.

Based on this evaluation, prior NRC approval of these changes is not required.

STD DEP 2.2-5, CRAC2 and MACCS2 Codes

Description

This departure includes the use of another accident analysis computer code known as MACCS2 (MELCOR Accident Consequence Code System, Version 2) for the analysis of site-specific characteristics in the offsite dose assessment for STP 3 & 4. The reference ABWR DCD references the use of CRAC2, and the FSAR analysis supplements the existing DCD analysis. Since approval of the DCD, offsite dose methodology and computer codes have been improved, with the MACCS2 code considered the best available code for performing offsite dose analysis. Therefore MACCS2 is being included along with CRAC2.

Evaluation Summary

This departure has been evaluated in accordance with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The change represents an improved methodology to better calculate potential offsite doses and has no adverse impact.

The NRC has approved the use of MACCS2 for this type of analysis (NUREG/CR-6613). This change to the DCD incorporates the latest accident analysis computer code along with site specific data. The results were compared to the generic results using CRAC2 previously approved by the NRC in the ABWR DCD and found to be bounded by the earlier criteria. The change in analysis methodology does not introduce new equipment nor does it affect redundancy. The site specific accident consequences reanalysis performed with MACCS2 demonstrates that the acceptance criteria have been met. No design basis limit for a fission product barrier is being exceeded or altered by this departure. There is no impact on any Tier 1, Tier 2*, Technical Specifications, Bases for the Technical Specifications or operational requirements as a result of this departure. Consequently, prior NRC approval is not required.

STP DEP 3.5-1, Missile Protection

Description

This departure addresses the change from a single unit with a favorable turbine generator placement and orientation to a dual unit in which the turbine generator placement and orientation is considered unfavorable to essential systems of the adjoining unit per Regulatory Guide 1.115. The probability for missile generation (P1) is revised accordingly and criteria for Licensee Actions contained in Table 3.5-1 are revised in accordance with Standard Review Plan 3.5.1.3. Figure 3.5-2 is revised to show the +/- 25 degree low-trajectory turbine missile ejection zones for the two adjacent STP 3 & 4 units and the relation to corresponding essential equipment of the adjoining unit.

Evaluation Summary

The NRC has provided guidance in Regulatory Guide 1.115, Rev 1 for protection against low-trajectory turbine missiles. Further criteria were provided in NUREG-0800, Standard Review Plan (SRP) Section 3.5.1.3. Previously the DCD contained information based on a single unit plant. This was evaluated in the FSER (NUREG-1503) Chapter 3.5.

The design in the DCD assumed a single unit with a favorably oriented turbine. The STP 3 & 4 dual unit plant has an unfavorably oriented turbine when considered in relation to the other unit. The SRP also addresses an unfavorable orientation. This change incorporates the probability values for an unfavorable orientation in Table 3.5-1.

The change does not affect Tier 1, Tier 2*, Tech Specs, the basis for Tech Specs or Operational Requirements. The NRC Commitment for a turbine system maintenance program is being tracked separately (COL Item 3.13, Commitment Number COM 3.5-1).

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. Although not part of a safety related system, the turbine is a potential source of high energy missiles that could damage SSCs important to safety. The turbine is designed to minimize the possibility of failure of a turbine blade or rotor. That design in addition to the recommended maintenance and inspection program ensures that the probabilities of missile generation are maintained at or below the acceptable level contained in the SRP. As a result, this change will not result in more than a minimal increase in the probability of an accident. The change does not affect the consequences of previously evaluated accidents. Turbine missiles have been addressed in the DCD. The change results in more targets but that change does not affect the possibility of a malfunction of any SSC important to safety and therefore does not result in an increase to the consequences of such a malfunction. This change results in no new accident scenarios which were not previously analyzed. This change does not affect the design basis limits for fission product barriers. The methodologies for evaluating turbine missiles are outlined in RG1.115 and SRP Section 3.5.1.3. This change does not depart from those methodologies.

The NRC established criteria for both favorably and unfavorably oriented turbines. While this change results in an unfavorable orientation and thus in an increase in the product of strike probability and damage probability, the missile generation probability values (P1) included in Table 3.5-1 are such that the probability calculation of unacceptable turbine missile damage results in a probability of less than $1x10^{-7}$ per year for STP 3 & 4 and therefore no new ex-vessel severe accidents will become credible. This calculation will be made available for NRC review (COL Item 3.13, Commitment number COM 3.5-1).

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 3.6-1, Main Steam Tunnel Concrete Thickness

Description

DCD Tier 2 Subsection 3.6.1.3.2.3 specifies a 2 meter minimum wall thickness for concrete in the main steam tunnel. However, a minimum concrete wall thickness of less than 2 meters is acceptable in some locations, based on structural and shielding calculations. The location-specific minimum required wall thickness will be evaluated. ABWR DCD Tier 2 Subsection 3.6.1.3.2.3 is revised to remove this requirement. STP 3 & 4 will be designed using the more general requirement specifying a Steam Tunnel thickness of 1600 mm or greater provided in ABWR DCD Tier 1 Table 2.15.12.

Evaluation Summary

The Steam Tunnel concrete thickness has been evaluated for structural stability and radiation shielding.

The structural calculation has been based on ACI 349 requirements using estimated loads from a similar plant.

The DCD specifies ACI 349-80 edition. However, based on a Code Edition Change, ACI 349-97 is used for this evaluation and is considered equivalent. The results of the structural calculation demonstrate that the concrete thicknesses below the ABWR DCD Tier 2 Subsection 3.6.1.3.2.3 requirements are adequate.

A dose rate calculation for the MS Tunnel has also been performed at the most critical radiation shielding requirement areas in the Reactor and Control Building. These calculations were performed using a N-16 activity concentration of 11.1 MBq/g, as specified in the DCD. The calculation shows that Steam Tunnel concrete thicknesses below the ABWR DCD Tier 2 Subsection 3.6.1.3.2.3 requirements are also adequate.

This proposed change affects the MS Tunnel concrete thickness, and does not affect any active plant SSC important to safety. Thus, there is no effect on any malfunctions previously evaluated in the DCD, therefore, there is no increase in occurrence and the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated.

The structural integrity and shielding requirement of MS Tunnel were verified. The proposed change is not relied upon for probability and consequences of an ex-vessel severe accident.

Consequently, the proposed change does not have any adverse impact on safety, accident and other aspects evaluated in the DCD, and prior NRC approval is not required for this departure.

STD DEP 3.8-1, Resizing the Radwaste Building

Description

Due to process changes to the radioactive waste treatment systems described in departures STD DEP 11.2-1 and 11.4-1, the dimensions and layout have changed from the DCD. The major changes from the Liquid Waste Management System (LWMS) and Solid Waste Management System (SWMS) described in the DCD are caused by the use of mobile processing equipment rather that permanently installed processing equipment. Permanently installed equipment is limited to the tanks and pumps used for collection, transfer and sampling of liquid and solid radwaste. In addition to the change to mobile processing equipment, some major system components described in the DCD, such as the radwaste evaporator, incinerator and compacter, will not be used. The result is a building that is slightly longer and slightly narrower than the Radwaste Building described in the DCD. The overall height of the new layout is smaller than the DCD layout, but the depth of the below grade substructure (which contains all of the liquid storage) is nearly identical to the below grade substructure described in the DCD.

The new layout of the Radwaste Building is depicted on the general arrangement drawings in FSAR Section 1.2 (Figures 1.2-23a through 1.2-23e). The new arrangement also affected the fire protection drawings in FSAR Section 9a (Figures 9A.4-28 through 9A.4-32). Note that although the fire protection drawings are affected, there is no effect on the Fire Hazards Analysis contained in FSAR Section 9A. As stated in FSAR Section 9A.4.5, there are no safe shutdown components located in the Radwaste Building, so there is no evaluation of fire hazards in specific rooms in the Radwaste Building. The new arrangement of the Radwaste Building, so there is no evaluation of fire hazards in specific rooms in the Radwaste Building. The new arrangment of the Radwaste Building also required that the radiation zone drawings and the radiation monitor location drawings be updated for the new layout (Figures 12.3-37 through 12.3-41, and Figures 12.3-65 through 12.3-68, and Table 12.3-6).

Evaluation Summary

This departure has been evaluated pursuant the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on Tier 1, Tier 2*, technical specifications, basis for technical specifications, or operational requirements as a result of this change.

This is a change to the dimensions and arrangement of the Radwaste Building. The limiting accident for the Radwaste Building is the failure of the Low Conductivity Waste (LCW) tank and the subsequent airborne release. The new arrangement is based on a different number of tanks with different sizes. However, this change does not alter the design of the radwaste storage tanks and therefore will not increase the frequency of a tank failure. The change in layout results in a footprint for the new building that is slightly larger than the Radwaste Building described in the DCD, and the depth of the below grade substructure is nearly identical to the Radwaste Building described in the DCD. The construction of the below grade substructures includes concrete slabs and walls that are lined with steel. The new layout results in all equipment (tanks) containing contaminated liquid located in the below grade substructure, similar to the Radwaste Building design in the DCD. Therefore, a release to groundwater caused by the new layout of the Radwaste Building is not considered credible and the airborne release remains bounding. Therefore, the proposed change does not result in more than a minimal increase in the frequency or the consequences of the limiting accident previously evaluated in the DCD.

No new processes or equipment are introduced by this change. Therefore, the proposed change does not result in more than a minimal increase in the occurrence or the consequences of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the DCD.

All equipment containing a large quantity of radioactive liquid is located in the substructure of the building so that failure of the equipment will lead to the collection of the liquid in the building substructure. This prevents liquid release to the groundwater or to the surface of the ground in the event of a tank failure. This design is consistent with the design of the Radwaste Building in the DCD. Therefore, the proposed change does not create the possibility for an accident of a different type than evaluated previously in the DCD.

The Radwaste Building does not contain any safe shutdown or other safety related equipment, so no equipment important to safety is affected by the new layout of the Radwaste Building. Therefore, the proposed changes do not create the possibility for a malfunction of an SSC important to safety with a different result than evaluated previously in the DCD.

The changes to the Radwaste Building dimensions do not involve any interaction with the fuel, reactor system boundary, or the containment boundary. Therefore the proposed change does not affect the fission product barriers as described in the DCD.

No evaluations related to the plant safety analysis are affected by the changes to the dimensions of the Radwaste Building. Therefore, the proposed change does not result in a departure from the method of evaluation described in the DCD used in establishing the design basis or in safety analysis.

The changes to the dimensions and arrangement of the Radwaste Building do not involve any interaction with fuel, reactor system boundary, or the containment structure or interact directly with systems associated with ex-vessel severe accidents or severe accident mitigation. Therefore, there is no substantial increase in the probability or consequences of an ex-vessel severe accident previously reviewed.

Based on this evaluation, prior NRC approval of the change in not required.

STD DEP 3.9-1 Reactor Internals Materials

Description

DCD Tier 2 Section 3.9.5.1.2.9 states that the reactor incore guide tubes have “two levels of stainless steel stabilizer.” It specifies the material as stainless steel for the stabilizer. On all currently operating ABWRs, the lower level of stabilizer is Ni-Cr-Fe Alloy. This departure specifies that there are two levels of Incore guide tube stabilizers. The upper stabilizer is welded to Shroud made from stainless steel. The lower stabilizer is welded to Shroud Support made from Ni-Cr-Fe alloy. The material of stabilizers needs to be the same or similar material as the components to be welded in order to minimize differential thermal expansion. Therefore, the upper stabilizer needs to be stainless steel and the lower stabilizer needs to be Ni-Cr-Fe Alloy.

Evaluation Summary

This change has been evaluated pursuant to the requirements in 10CFR52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, DCD, technical specifications, basis for technical specifications or operational requirements as a result of this change.

Reliability of the Incore Guide Tube Stabilizer is increased using the same material as the welded component and minimizing the differential thermal expansion. Due to the increased reliability of the Incore Guide Tube stabilizer provided by this change, this change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the DCD. This change does not affect any systems relied upon to prevent or mitigate a severe accident.

Based on this evaluation, prior NRC approval of the change is not required.

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

Description

Reference ABWR DCD Appendix 3B, Section 3B.4.2.3 provides two equations for calculating the pulse duration for a flat target, one of which, for V < 2.13 m/s, is:

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

Where:

T= the duration of impact (seconds)

W= the width of the flat structure (meters)

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

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, technical specifications, bases for technical specifications or operational requirements as a result of these changes.

The change affects a multiplying factor required for the correct application of units. It does not affect the design or function of the structures and components that can be impacted by a suppression pool swell. The correct loads are used for the structural analyses to show that the structures and components withstand the loads adequately and no failure results. The change does not apply to the analyses of containment penetrations or containment boundary. Consequently, there is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety.

This change involves a correction of a formula, but does not alter the hydrodynamic method of evaluation for the pool swell load effects, or any method used in the design bases or safety analyses. The structures and components above the pool have not been identified as a design feature in the plant specific DCD for mitigating an ex-vessel severe accident. Therefore, the likelihood or consequences of a severe accident is not impacted.

As a result of this evaluation, prior NRC approval of the change is not required.

STD DEP 3H -1, Liner Anchor Material

Description

ABWR DCD Tier 2 Subsection 3H.1.4.4.3 incorrectly identifies the Containment Liner Anchor material as ASTM A-633 Gr. C, which is not an ASME Code allowable material. ASTM A-633 Gr. C is inconsistent with ABWR DCD Subsection 19F.3.2.1, which identifies the Containment Liner Anchor Material used in the containment severe accident evaluation as ASTM A-36. ASME SA-36 and ASTM A-36 have the same physical properties. This departure corrects the Containment Liner Anchor material identified in Subsection 3H.1.4.4.3 to SA-36.

Evaluation Summary

This change has been evaluated pursuant to the requirements set forth in 10 CFR 52 Appendix A Section VIII.B.5. This change does not affect Tier 1, Tier 2* or Technical Specifications, the bases for Technical Specifications, or operational requirements. The containment severe accident structural evaluation described in ABWR DCD Tier 2 Chapter 19F includes an assessment of the Containment Liner and Liner Anchors. The material properties used for the Liner Anchors in the Subsection 19F.3.2.1 analysis are for ASTM A-36. ASME permitted material SA-36 meets the requirements of ASTM A-36. This evaluation, supplemented with Bechtel Topical Report BC-TOP-1, “Containment Building Liner Plate Design Report” (ABWR DCD Reference 19F-7), demonstrates that the A-36 Liner Anchor material is acceptable. Furthermore, ABWR

DCD Tier 2 Subsection 3H.1.5.1 indicates that the Liner Anchors are considered rigid links and are not explicitly evaluated in the containment analysis. Based on this, the use of SA-36 Containment Liner Anchors is appropriate and ABWR DCD 3H.1.4.4.3, which identifies the material as ASTM A-633 Gr. C, is incorrect. Subsection 3H.1.4.4.3 is revised to identify the material as SA-36.

No change is required for Section 19F.3.2.1 because this section is documenting an historical calculation and the properties of ASME SA-36 and ASTM A-36 are the same.

This departure does not change any severe accident evaluation including those in ABWR DCD Tier 2 Chapter 19F. Therefore, the likelihood of a severe accident is not impacted. There is also no impact on the probability or consequences of an accident or malfunction of an SSC important to safety. Furthermore, there is no impact on fission product barriers. Therefore, this change has no adverse impacts and does not require prior NRC approval.

STD DEP 3I-2, Environmental Qualification - Radiation

Description

This departure revises the integrated gamma radiation dose for the main steam tunnel presented in Table 3I-17. The increase in this value is based on current results of post-accident radiation calculations and analysis. These results show increases in the integrated accident gamma dose to the affected area. Table 3I-17 was updated to ensure that equipment located in this area will meet their design requirements to operate in a post-accident environment. Therefore, this change ensures continued compliance with the regulatory requirements for safety-related equipment.

Evaluation Summary

This departure is the result of re-evaluation of the post-accident radiation conditions inside the reactor building. There is no impact on the frequency of occurrence or the consequences of an accident as a result of this change. In addition, any SSCs important to safety will be qualified to the revised radiation dose limit, so there is no impact on the likelihood of occurrence or consequences of a malfunction of an SSC important to safety as a result of this change.

There is no new accident scenario or malfunction of an SSC important to safety as a result of this change. A design basis limit for a fission product barrier is not exceeded or altered. A method of evaluation in establishing the design bases or in the safety analyses does not change. There is no impact on the probability or the consequences to the public of an ex-vessel severe accident.

Based on this evaluation, prior NRC approval of these changes is not required.

STD DEP 3MA-1, Correction of Inconsistencies In System Evaluation for ISLOCA

Description

The system evaluation for ISLOCA as described in the ABWR DCD Tier 2 Appendix 3MA. This departure consists of the following correction of inconsistencies between Appendix 3MA and P&IDs in Chapter 21.

The following corrections are made:

(1) Addition of missing components.

(2) Correction of nominal diameter of the piping.

(3) Correction or addition of the P&ID sheet number.

(4) Correction of the number of a component.

(5) Correction of design pressure of a component. No change to the actual design pressure of a system.

(6) Correction of design temperature of a component. No change to the actual design temperature.

(7) Correction of group classification of a component.

(8) Deletion of an unnecessary component.

(9) Correction of seismic category of a component.

(10) Correction of a typographical error.

Evaluation Summary

This departure has been evaluated in accordance with 10 CFR 52, Appendix A, Section VIII.B.5. This change is a correction of inconsistencies. The correction of the ISLOCA evaluation does not adversely impact the ABWR design. In fact, these clarifications add numerous valves and piping to the list of ISLOCA upgraded components, thus providing greater protection against ISLOCA events, thereby, reducing the probability and consequence of accidents and failure of SSCs important to safety.

Consequently, prior NRC approval is not required.

STD DEP 4.5-1, Reactor Materials

Description

The description of the materials for the control rod drive (CRD) mechanisms in Section 4.5.1 and the reactor internals in Section 4.5.2 of the DCD has been revised (1) to reflect the materials successfully used in operating ABWR designs over the last 10 years; (2) to clarify some data and provide equivalent materials, as appropriate; and (3) to clarify some fabrication and material issues for reactor internals materials.

The description of Code Case applied to RPV, Reactor Internals has been revised in Section 5.2 to reflect the issuance of “N-580-2”.

A summary of the changes to the DCD is as follows:

(1) All components made from austenitic stainless steels in the original DCD material listings for the CRD mechanisms and the reactor internals are still fabricated from austenitic stainless steels in the revised listings. For example, 316 stainless steel is added to the original DCD material listings.

(2) Where equivalent materials are now listed for components, the equivalent material has demonstrated successful application and operation with no impact on the design and safety function of the CRD mechanisms and reactor internals.

(3) In some cases, wrought material is specified instead of cast, or the product form specification has been changed.

(4) The description of the experience base for the materials has been updated to no longer exclude Type XM-19 stainless steel and now covers 25 years.

(5) For components that are reactor pressure boundary components, the materials meet ASME Code, Section III, Class 1 or Class CS requirements (Subsection NB or NG).

(6) The paragraph discussing nondestructive examination (NDE) of wrought seamless tubular products was revised to delete reference to the peripheral fuel supports, which are not tubular products, and to address the fact that the CRD housings are reactor coolant pressure boundary components (ASME Code, Section III, Class 1) as well as core support structures (Class CS).

(7) The paragraph discussing controls on welding was clarified to make it clear industry welding standards are applicable to internals other than core support structures and that qualification per Section IX of the ASME Code applies to the core support structures.

(8) The paragraph discussing the fabrication and processing of austenitic stainless steel was revised to clarify the limitations on delta ferrite content in weld materials and castings. This paragraph was also revised to more clearly state how the ABWR will comply with the intent of Regulatory Guide 1.44.

(9) Mechanical properties of Alloy X-750 and heat treatment method are changed. Alloy X-750 with changed mechanical properties is used for latch and bayonet coupling materials. The material has demonstrated successful application and operation with no impact on the design and safety function of the CRD

mechanisms by the Toshiba reliability verification testing. Furthermore, the DCD is changed to reflect the fact that this material has been used in nuclear power plants for more than 25 years.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1 or Tier 2* DCD, Technical Specifications, Basis for Technical Specifications or operational requirements as a result of these changes.

The changes associated with this departure are primarily equivalent material substitutions, changes in material form, editorial clarifications and format changes. Since the proposed material or editorial changes are equivalent to the original DCD design, there is no effect on any accident previously evaluated in the plant specific DCD. Furthermore, it doesn't change any plant physical features, SSCs important to safety, or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not have an adverse impact on any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, the change has no adverse impact and does not require prior NRC approval.

STD DEP 4.6-1, FMCRD Friction Test Equipment

Description

ABWR Tier 2 Subsection 4.6.1.2.3(5) describes a test fixture used in Fine Motion Control Rod Drive (FMCRD) friction testing. The test fixture contains a small pump and associated hydraulic controls to pressurize the underside of the hollow piston of FMCRD. The proposed departure removes the small pump from the test fixture. Water for friction testing is supplied from the CRD pump discharge.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10CFR52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this departure.

The proposed departure removes the small pump from the test fixture for use in FMCRD friction testing. Water for the test fixture is supplied from the CRD pump discharge. This test fixture is only used for testing and not during reactor operation. The change has no impact on the FMCRD or Hydraulic Control Unit (HCU) component design or function. Furthermore, this departure has no impact on any SSC important to safety, and it does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the DCD. The FMCRD friction test equipment has not been identified as a design feature in the DCD for mitigating an ex-vessel severe accident. Fission product barriers are not impacted by the proposed departure.

Based on this evaluation, prior NRC approval of the change is not required.

STD DEP 5.2-2, PSI/ISI NDE of the Reactor Coolant Pressure Boundary

Description

A departure from DCD Subsections 5.2.4.2.2 and 5.2.4.3.1 is provided for PSI and ISI of welds in Reactor Coolant System piping to meet the requirements of ASME Section XI, Appendix VIII as mandated by 10 CFR 50.55a, rather than meeting the requirements of Regulatory Guide 1.150, Rev. 1.

Evaluation Summary

This departure has been evaluated pursuant to 10 CFR 52, Appendix A, Section VIII.B.5.

This change is necessary because of the NRC withdrawal of the previous Regulatory Guide following the requirements being codified in 10CFR50.55a (g)(6)(ii)(C)(1). It provides additional guidance and does not adversely affect any functional or safety requirements. Since this change does not affect any other plant SSCs, there is no effect on any accident previously evaluated in the DCD. This departure does not change the Technical Specifications or any other underlying design. The operational requirements for conduct of preservice inspection remain unchanged. This departure does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 5.3-1, Reactor Pressure Vessel Material Surveillance Program

Description

This departure pertains to the RPV material surveillance program. The RPV material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation. The test results of the RPV material surveillance program are used to evaluate pressure and temperature limitations for the RPV hydro pressure test. This departure is a clarification of the number of the test specimens, the lead factor, and it addresses COL License information Item 5.5 of the DCD. Specifically.

(1) ABWR DCD Tier 2 Subsection 5.3.1.6.1 specifies the number of the test specimens required by ASTM E185-82 for each specimen capsule. This departure clarifies that the number of test specimens listed in the Reference DCD are the minimum amount required by the Standard. Additional test specimens may be included in the capsules.

(2) ABWR DCD Tier 2 Subsection 5.3.1.6.4 specifies that the applicable lead factor for surveillance capsule is approximately 1.2 to 1.5. On all currently operating ABWRs, the capsules are placed closer to the vessel wall and the lead factors are approximately 1.1. Therefore, this departure clarifies the width of the lead factors are 1 to 1.5.

(3) A reference for the RPV Material Surveillance Program for STP 3&4 is added in DCD Subsection 5.3.5.

Evaluation Summary

This change has been evaluated pursuant to the requirements in 10CFR52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change. This changes the RPV Material and Surveillance program only; there are no changes to any plant systems or to the manner in which the plant is operated. This program ensures that the RPV maintains its fracture toughness margins throughout the vessel lifetime.

Therefore, this change does not result in more than a minimal increase in the frequency of occurrence or consequences of an accident previously evaluated in the DCD, and it does not increase the likelihood or consequences of malfunctions previously evaluated, does not effect design basis limits for fission product barriers, and does not involve a method of evaluation.

This change does not affect any systems and therefore does not affect the evaluation of ex vessel severe accidents.

Based on this evaluation, prior NRC approval of the change is not required.

STD DEP 5.4-1, Reactor Water Cleanup System

Description

The flow capacity of the two pumps and two filter demineralizers in the Reactor Water Cleanup System are doubled from 1% of rated feedwater flow to 2%. This will improve system maintainability and availability by allowing only one of the two pumps to handle the full cleanup flow and filtering requirements. The pump discharge head at shutoff is increased from 160m to 182m and the design pressure of the pumps and heat exchangers is changed from 10.20 MPaG to 10.65 MPaG as described in Table 5.4-6 of the FSAR.

Evaluation Summary

This departure has been evaluated in accordance with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on Tier 1, Tier 2*, technical specifications, basis for technical specifications and operational requirements.

The proposed change improves CUW system reliability by providing backup pump and filter demineralizer capability to handle 100% CUW flow and filtering requirements. The probability of situations whereby system flow would be greater than 100% (such as operating both pumps at the same time) are minimized since the CUW system flow rate is controlled to its rated flow by the flow control valve at the outlet of the filter demineralizer when two CUW pumps run at the same time for some duration, as described in the DCD. Consequently, the effect on accident and SSC malfunction frequency is minimal.

A break in the CUW system, should one occur, will be automatically isolated by either of two redundant safety related isolation valves which receive signals including low reactor water level, high ambient CUW equipment room area temperature, high main steam tunnel area temperature and high mass differential flow. Any one of these signals will isolate the system. All of the piping upstream of the two safety related isolation valves is Seismic Category I and Quality Group A and is located inside the containment. All CUW operating conditions, including pump flow under normal operating conditions are unchanged from the DCD. Consequently, there is no adverse impact on accident or SSC malfunction consequences previously analyzed nor is there an adverse affect on likelihood or consequences of a severe accident.

There are no new accident scenarios as a result of this change. Any breaks in the CUW system, should they occur, would continue to be bounded by the feedwater line break.

As a result, this departure satisfies all VIII.B.5 criteria and prior NRC review is not required.

STD DEP 5.4-2, Reactor Internal Pump (RIP) Motor Cable Box

Description

Subsection 5.4.1 of the reference ABWR DCD describes component and subsystem design information of the Reactor Recirculation System. The FSAR revises the RIP cross section illustration, Figure 5.4-1, to reduce the size of the cable box and to show a plug-in type power connector. Neither the motor cable box nor the power connector is described in any section of the DCD.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, DCD technical specifications, basis for technical specifications or operational requirements as a result of this change.

This change revises the RIP motor cable box to a smaller size box with a plug-in power connector. It has no effect on the Reactor Internal Pump operation or performance. The change is made to improve maintainability. The RIP motor cable box and plug-in connector are nonsafety-related components that are not required for safe shutdown or accident mitigation, and thus this change does not increase the probability or consequences of any previously analyzed accident or malfunction of an SSC important to safety, and does not create the possibility of a different type of accident than previously analyzed or the possibility of a malfunction of an SSC with a different result. It has no effect on fission product barrier limits, and does not involve a change in methodology.

The RIPs are not design features important for ex-vessel severe accidents. Therefore this change has no impact on ex-vessel evaluations.

Based on this evaluation, prior NRC approval of these changes is not required.

STD DEP 5.4-3 Residual Heat Removal System Interlock

Description

ABWR DCD Tier 2 subsection 5.4.7.1.1.6 states that the wetwell spray can be used in conjunction with the low pressure flooder (LPFL) mode. However, ABWR DCD Tier 2 Figure 7.3-4 sheet 11 of 20 indicates that the wetwell spray valve will close when the LPFL injection valve is not fully closed. This is in accordance with the ABWR design and the assumptions of the safety analysis. Therefore, the statement in the DCD that the wetwell spray can be operated in conjunction with the LPFL mode is inconsistent and clarified by the proposed departure.

ABWR DCD Tier 2 Table 5.4-3 NOTE C indicates that Minimum Flow Valves open logic is “Pump is running and low loop flow signal.” However, ABWR DCD Tier 2 Figure 7.3-4 sheet 12 of 20 indicates that Minimum Flow Valves open logic is “Pump discharge pressure high and low loop flow signal.” This proposed departure clarifies the Minimum Flow Valves open logic in Table 5.4-3 to be consistent with the figures.

ABWR DCD Tier 2 Table 5.4-5 indicates that relief pressure of E11-F028A-C and E11-F051A-C is 3.44 MPaG. However, ABWR DCD Tier 2 Figures 5.4-10 sheets 3, 4 and 6 indicate that design pressure of these relief valves is 3.43 MPaG. This proposed departure clarifies the relief pressure of E11-F028A-C and E11-F051A-C in Table 5.4-5 to be consistent with the figures.

Evaluation Summary

This departure has been evaluated in accordance with 10 CFR 52, Appendix A, Section VIII.B.5. The proposed departures are corrections of inconsistencies in the ABWR DCD Tier 2 information. The proposed departures make no change to the ABWR design. Consequently, prior NRC approval is not required.

STD DEP 5.4-4 Recirculation Motor Cooling System

Description

ABWR DCD Tier 2 Section 5.4.1.3.1 identifies the Recirculation Motor Heat Exchanger (RMHX) shell, tube, sheet and water box material as carbon steel. This departure permits fabrication of these components using carbon steel or stainless steel. Similar changes are made to ABWR DCD Tier 2 Figure 5.4-4 for consistency.

Evaluation Summary

This departure has been evaluated in accordance with the requirements of 10 CFR 52 Appendix A, Section VIII.B.5. The departure has no impact on ABWR DCD Tier 1, Tier 2*, Technical Specification or Technical Specification Bases sections.

The proposed change offers enhanced resistance for RMHX components which are sensitive to flow assisted corrosion (FAC). ASME Boiler and Pressure Vessel Code structural margins are maintained. Accordingly, the departure results in less likelihood of a malfunction of any SSC important to safety. It also has no impact on the frequency of occurrence or consequences of an accident or ex-vessel severe accident previously evaluated. There is no impact on design basis limits to fission product barriers. Thus, the departure meets the requirements outlined in 10 CFR 52, Appendix A Section VIII-B.5, and prior NRC review is not required.

STD DEP 5.4-5 Addition of a vent line from the Reactor Water Cleanup System Reactor Pressure Vessel (RPV) head-spray line to the Reactor Head Vent Line.

Description

Addition of a vent line for RPV head-spray line to the Reactor Water Cleanup System (CUW) is to preclude accumulation of hydrogen gas in the head spray line. The vent line will connect to the RPV head vent line. ABWR DCD Tier 2 Figure 5.4-12, Reactor Water Cleanup System P&ID, and Figure 5.1-3 Nuclear Boiler System P&ID will be changed as a Standard Departure.

The hydrogen gas accumulation problem was identified at Hamaoka #1 (H-1) and Brunsbuttel BWRs. As a result, the NRC initiated GI-195 "Hydrogen Combustion in Foreign BWR Piping" and formulated a plan of action. The result was an analysis of core damage probability caused by piping rupture consequent to hydrogen combustion, modeled for the Peach Bottom plant. The probability of the event was evaluated at 10-7.

The RPV head-spray line has a "high-point" for injection of spray water from the RPV upper elevation. Since the head-spray piping contains no fluid during normal operating conditions, the potential for accumulating hydrogen gas exists. It is difficult to remove the high-point in the piping design; therefore adding a vent line is the optimum solution. The vent line connects to the RPV vent (suction) line to avoid hydrogen gas accumulation.

Evaluation Summary

This departure has been evaluated pursuant to the requirement in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on Tier 1, Tier 2*, or Technical Specifications. The changes have no effect on the frequency or consequences of accidents, or the probability or consequences of malfunctions. The systems involved are not relied upon for ex-vessel severe accident mitigation. This change is an improvement in safety, designed to reduce the probability for accidents.

As a result, prior NRC approval of this change is not required.

STD DEP 5A-1, Delete Appendix on Compliance with Regulatory Guide 1.150

Description

NRC requirements for performance demonstration of ultrasonic examination of reactor pressurevessel and piping for preservice inspection and inservice inspection once addressed by RG 1.150 will be conducted in accordance with ASME Section XI, Appendix VIII as required by 10 CFR 50.55a.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5

RG 1.150 was issued by the NRC in the early 1980’s in order to provide guidance on ultrasonic testing of reactor vessel welds during preservice and inservice examanations at a time when the Staff felt that industry guidance was inadequate. The NRC subsequently withdrew the RG in February, 2008 and in its withdrawal stated that the requirements for such testing have been superceded by 10 CFR 50.55a. Specifically, 10 CFR 50.55a requires both preservice and inservice inspection activities to be performed using personnel, equipment, and procedures qualified in accordance with the ASME, Boiler and Pressure Vessel Code, Section XI, Appendix VIII. This requirement is met by STP Units 3 & 4 as stated in Subsection 5.2.4 of the STP 3 & 4 FSAR. With the withdrawal of RG 1.150, no evaluation against that RG is required and the plant meets the latest NRC requirements. Consequently, this change has no impact on the likelihood or consequences of an accident or malfunction of an SSC important to safety previously evaluated. There is no impact on ex-vessel severe accident likelihood or consequences.

This change has no impact on any Tier 1, Tier 2*, Technical Specifications, bases for the Technical Specification, or operational requirements information.

Based on the results of this evaluation, prior NRC approval of this change is not required.

STD DEP 5B-1, Residual Heat Removal Flow and Heat Capacity Analysis

Description

The K-Value* for the Residual Heat Removal heat exchangers is increased from 3.69×10^5 W/°C to 4.27×10^5 W/ C to reflect an ultimate heat sink water temperature of 35°C . The limiting event for heat exchanger sizing is now the rapid cooldown required for a 17-day outage in accordance with the Utilities Requirement Document. Previously, the limiting event for heat exchanger sizing was a LOCA.

This departure increases the heat removal capacity of the RHR heat exchanges to allow reduced outage time. This change improves system performance, maintainability, and availability.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, as described previously. The proposed change improves the heat removal capability of the RHR system for off-normal and accident events by specifying a more conservative design condition for sizing of the RHR heat exchangers.

This change will not impact the reactor systems in any way which would result in any increase in the likelihood of an accident or a malfunction of an SSC. In addition, the consequences of an accident or malfunction of an SSC with implementation of this change will not be affected.

There are no new accident scenarios or SSC failures leading to a different result as presented in the DCD as a result of this change. By designing the RHR heat exchangers to a higher heat removal capacity, the heat removal capability of the RHR system is enhanced and additional heat removal capability margin is being added relative to the DCD. The change will not adversely impact the performance or capability of any fission product barriers.

The method of evaluation being used to size the heat exchangers is consistent with the method provided in Appendix 5B to Part 2 Tier 2 of the DCD. Neither the likelihood nor consequences of an ex-vessel severe accident are adversely affected by this change. The increase in the heat removal capability of the RHR heat exchangers improves mitigating the consequences of a severe accident. Consequently, prior NRC approval is not required.

* The K-Value, or K-Factor, is a method of determining the amount of heat transferred per unit of time based on the temperature difference across the heat exchanger without considering the heat exchanger area or other heat exchanger factors.

STD DEP 6.2-3, Containment Penetrations and Isolation

Description

This departure corrects primary containment penetration errors and inconsistencies in Section 6.2 of the reference ABWR DCD and provides additional design detail that was not present in the reference ABWR DCD. This departure is the result of detailed 3-D layout analysis that was performed to ensure that the penetrations meet U.S. codes and standards for mechanical and electrical separation. Changes to the tables include the correction of containment penetration elevation, azimuth, offset, and diameter. In addition, containment isolation barrier type information is provided for valves that did not contain this level of detail in the reference ABWR DCD.

This departure primarily affects the detailed containment isolation valve listings in Tables 6.2-5, 6.2-6, 6.2-7, 6.2-8, and 6.2-10. The changes to each table are discussed below. Because Tables 6.2-7, 6.2-8, and 6.2-10 are also affected by four other departures, listings of the penetrations that are changed due to STD DEP 6.2-3 are provided .

- Table 6.2-5 identifies the reactor coolant pressure boundary (RCPB) influent lines penetrating the drywell. This change consists of the correction of inconsistencies of valve type between Table 6.2-5 and Chapter 21 P&ID.
- Table 6.2-6 identifies the RCPB effluent lines penetrating the drywell. This change consists of the correction of inconsistencies of valve type between Table 6.2-6 and Chapter 21 P&ID.
- Table 6.2-7 identifies the containment isolation valves associated with the ABWR Containment. This change consists of the following correction of inconsistencies between Table 6.2-7 and drawings of Chapter 21.

  (1) Addition of the missing valve that should be listed.

  (2) Correction of line size.

  (3) Correction of valve type.

  (4) Clarification of valve position in various operating condition.

  (5) Correction of note.

  (6) Correction of a typographical error.

  (7) Clarification of applied GDC.

  (8) Correction of power source (division). (To be consistent with Tier 1 Figures 2.1.2b and 2.11.5.)

  (9) Change of closure time of T31-F009 to be identical with T31-F005 for bleed isolation.

In Table 6.2-7, the following are changed:

- Line size for Reactor Recirculation System Valves B31- F008A- H/J/K
- Leak test type for Standby Liquid Control System Valves C41-F008 and F006A/B
- Valve position for Containment Atmospheric Monitoring System valves D23-F004A/B, F005A/B, F006A/B, F007A/B and F008A/B
- CIV signal for Containment Atmospheric Monitoring System valves D23-F001A/B, F004A/B, F005A/B, F006A/B, F007A/B and F008A/B
- Valve position for Residual Heat Removal System Wetwell Spray Valves E11-F019B/C
- Valve position for Residual Heat Removal System Drywell Spray Valves E11-F017B/C and F018B/C
- Valve position for Residual Heat Removal System Minimum Flow Line Valves E11-F021A/B/C
- Valve position for Residual Heat Removal System S/P Suction (LPFL)Valves E11-F001A/B/C
- Valve position for Residual Heat Removal System Inboard Shutdown Cooling E11-F010A/B/C
- Valve position for Residual Heat Removal System Outboard Shutdown Cooling E11-F011A/B/C
- Valve position for Residual Heat Removal System Injection and Testable Check Valves E11-F005B/C and F006B/C
- Valve position and CIV signal for High Pressure Core Flooder System S/P Suction Valves E22-F006B/C
- CIV signal for High Pressure Core Flooder System Test and Minimum Flow Valves E22-F009B/C and F010B/C
- Valve position for High Pressure Core Flooder System Injection Valves E22-F003B/C and F004B/C
- Primary actuation for Nuclear Boiler System Main Steam Lines A, B, C and D Valves B21-F009 A/B C/D
- Valve position for Nuclear Boiler System Feed Water Line A and B Valves B21-F004 A/B and B21-F003 A/B

- Valve position for Reactor Core Isolation Cooling System Steam Supply Valves E51-F035, F048, and F036
- Valve position for Reactor Core Isolation Cooling System S/P Suction Valve E51-F006
- Valve position for Reactor Core Isolation Cooling System Turbine exhaust Valve E51-F038
- CIV signal for Atmospheric Control System Valves T31-F001, F002, F003, F004, F005, F006, F008, F009, F025, F039, F040, F041 and F011
- Closure time for Atmospheric Control System Valve T31-F009
- CIV signal for Reactor Water Cleanup System Valves F002, F003 and F017
- Valve information for Reactor Water Cleanup System valves G31-F071 and F072
- Leak test type, Valve position and CIV signal for Suppression Pool Cleanup System Valves G51-F001, F002, F006, and F007
- GDC basis and Valve position for Reactor Building Cooling Water System Valves P21-F075A/B, F076A/B, F080A/B, F081A/B
- GDC basis for HVAC Normal Cooling Water System Valves P24-F053, F054, F142 and F141
- CIV signal for HVAC Normal Cooling Water System Valves P24-F053, F142 and F141
- Power source for HVAC Normal Cooling Water System Valve F141
- GDC basis for Instrument Air System Valves P52-F276 and F277
- GDC basis for High Pressure Nitrogen Gas Supply System Valves P54-F007A/B, F008A/B and F200/F209
- CIV signal for High Pressure Nitrogen Gas Supply System Valves P54-F007A/B, F008A/B and F200/F209
- CIV signal for Leak Detection & Isolation System Valves E31-F002, F003, F004 and F005
- GDC basis and CIV signal for Radwaste System Valves K17-F003, F004, F103 and F104
- Valve information for Neutron Monitoring System Valves C51-XXX A/B/C and XXX

Table 6.2-8 identifies the ABWR Primary Containment Penetrations. It was determined that this arrangement needed to be changed to meet US mechanical and electrical separation requirements. Containment penetrations are physically isolated in accordance with the requirement specified in 10CFR50, Appendix J. Containment electrical penetrations are physically separated in accordance with the requirement specified in Section 6.5 of IEEE Std 384-1992, which is endorsed by Regulatory Guide 1.75 Revision 3. Changes were also necessary to satisfy electrical load and to reflect the detailed physical location plan.

The following parameters are changed in Table 6.2-8 as a result of Departure STD DEP 6.2-3.

Penetration Elevation

Penetrations 37, 61, 62, 70, 92, 103 A/B/C/D/E, 110, 111, 112, 140A, 144A/B/C/D, 161A/B, 171, 201, 202, 203, 204, 205, 206, 213, 250, 321 A/B, 322 E/F, 323C, 331 A/B, 332 A/B, 600A/B/C/D, 610, 620,621, 650A/B/C/D, 651A/B/C/D, 680A/B, 700A/B/C/D/E/F/G/H/J/K, 710, 740, 750 A/B/C/D, 751 A/B/C/D, 780 A/B

Azimuth

Penetrations 90, 101 D/E/F, 102 F, 103 A/B/C/D/E, 104 E/G, 110, 111, 112, 160, 161 A/B, 171, 204, 205, 206, 241, 250, 321 A/B, 322 E/F, 323 C, 331 A/B, 332 A/B

Offset

Penetrations 92, 100B, 100E, 101 B/C/D, 101 C/G/F, 102 B/C/D/G, 103 A/B/C/D/E, 104 C/D/E/G, 105 C/D, 110, 111,112, 140 A, 161 A, 171, 250, 600 A/C/D, 610, 620, 621, 650 A/B/C/D, 651 A/B/C/D, 680 A/B, 700 A/B/C/D/E/F/G/H/J/K, 710, 740, 750 A/B/C/D, 751A/B/C/D, 780 A/B

Diameter

Penetrations 5, 6, 91, 92, 140 A, 100C, 101A/B, 102B, 103A/B/C/D, 104C/D/F/H, 105A/B/C/D, 147, 171, 250, 600A/B/C/D, 610, 620, 650 A/B/C/D, 651 A/B/C/D, 680 A/B, 700 A/B/C/D/E/F/G/H/J/K, 710, 750 A/B/C/D, 751 A/B/C/D, 780A/B

Barrier Type

Penetrations 10 A/B/C/D , 11,12 A/B, 22, 30 B/C, 31 A/B, 32 A/B, 33 A/B/C, 37, 38, 50, 60, 61, 62, 63, 64, 65, 66, 69, 70, 71 A/B, 72, 80, 81, 82, 90, 91, 92, 93, 110, 130 A/B/C/D, 140 A/B, 141 A/B, 142 A/B/C/D, 143 A/B/C/D, 144 A/B/C/D, 146 A/B/C/D, 147, 160, 161A/B, 162A/B, 170, 171, 177, 200 B/C, 201, 202, 203, 204, 205, 206, 210, 211, 213, 214, 216, 217, 240, 241, 242, 250, 252, 254, 320, 321A/B, 322A/B/C/D/E/F, 323A/B/C/D/E/F, 331A/B, 332A/B, 342, 600 A/B/C/D, 610, 650 A/B/C/D, 651 A/B/C/D, 680A, 700 A/B/C/D/E/F/G/H/J/K, 710, 740, 750 A/B/C/D, 751 A/B/C/D, 780A/B

Testing Type

10 A/B/C/D, 11,12 A/B, 37, 38, 50, 60, 65, 66, 69, 70, 80, 81, 90, 91, 92, 93, 110, 130 A/B/C/D, 140 A/B, 141 A/B, 142 A/B/C/D, 143 A/B/C/D, 144 A/B/C/D , 146 A/B/C/D, 147, 160, 161 A/B, 162A/B, 170, 171, 177, 213, 240, 241, 250, 252, 254, 320, 321A/B, 322A/B/C/D/E/F, 323 A/B/C/D/E/F, 331 A/B, 332A/B, 342, 650 A/B/C/D, 651 A/B/C/D, 680 A, 750 A, 780A, 780B, 750 B/C/D, 751 A/B/C/D

New

Penetrations 94, 95, 100F, 101J/K, 102 H/J, 106 A/B/C/D/F/G/H/J, 107A/B

Removed

Penetrations 113, 215, 220, 251, 253, 255, 300A/B

Table 6.2-10 contains the potential leakage paths from the Primary Containment to the environment. Corrections to this table included fields that were identified as requiring change based on changes to Table 6.2-8.

Penetration Diameter

Penetrations 5, 6, 91, 92, 100C, 101A/B, 102A/B, 103A/B/C, 104C/D/F/H, 105A/B/C/D, 110,140A, 171, 250, 252, 600 A/B/C/D, 610, 620, 650A, 650 B/C/D, 651 A/B/C/D, 680 A/B, 700 A/B/C/D/E/F/G/H/J/K, 750 A/B/C/D, 751 A/B/C/D, 780 A/B

Termination Region

Penetration 250

Leakage Barriers

Penetration 250

Potential Bypass Paths

Penetration 250

Added

Penetrations 94, 95, 100F, 101 J/K, 102 H/J, 103 D/E, 106 A/B/C/D/F/G/H/J, 107 A/B, 710, 740

Removed

Penetrations 113, 220, 215, 251, 253, 255, 300A/B, 334 , 341, 660D

Evaluation Summary

This change ensures the ABWR design conforms to US Codes and Standards, corrects errors and inconsistencies in the reference ABWR DCD, revises penetration locations to ensure they meet separation criteria based on 3-D layout analysis, and provides additional design information regarding containment isolation valve testing that was not present in the reference ABWR DCD. These changes collectively ensure the design is in full compliance with NRC rules and regulations and therefore do not impact the probability of occurrence of accidents, the consequence of accidents, and do not create accidents of a different type than previously evaluated. These changes do not adversely affect the containment fission product barrier, there is no change in any method of analysis, and there is no adverse effect on Severe Accident mitigation.

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The change has no adverse impact and prior NRC review and approval is not needed.

STD DEP 6.6-1, Pre-Service and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

Description

Departures from Subsections 6.6.2.1 and 6.6.2.2 of the reference ABWR DCD:

- A sentence in Subsection 6.6.2.1 regarding RHR heat exchangers nozzle having 100% accessibility for PSI during fabrication is deleted because it is no longer applicable.
- A paragraph in subsection 6.6.2.2 indicates restrictions for the use of some piping system configurations to ensure that accessibility for ISI is maintained. However, if some of the restricted piping system configurations are used, an evaluation is required to ensure ISI accessibility is provided.
- A sentence is added for clarification at the end of the Subsection 6.6.2.2 requiring an evaluation to be performed where less than the minimum straight pipe is used.
- The comprehensive plant-specific PSI and ISI program plan will be developed and submitted at least 12 months prior to commercial power operation.
- Access requirements are incorporated in the applicable specifications as an integral part of the design process.

Evaluation Summary

This departure has been evaluated pursuant to 10 CFR 52, Appendix A, Section VIII.B.5.

This change provides additional guidance and does not adversely affect any functional or safety requirements. Since this change does not affect any other plant SSCs, there is no effect on any accident previously evaluated in the plant-specific DCD. This departure does not change the Technical Specifications or any other underlying design. The operational requirements for conduct of preservice inspection remain unchanged. This departure does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

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

Description

A departure from Appendix 6C incorporates the new- complex ECCS strainers (e.g. Cassette Type Strainer) design per NUREG/CR-6224, NUREG/CR-6808 and from Utility Resolution Guidance for ECCS Strainer Blockage, NEDO-32868-A. The ECCS Strainer design also affects the description and the available NPSH of the ECCS pumps. Changes are made to DCD Tier 2 Section 5.4.7.2.2 and Tables 5.4-1a, 5.4-2, 6.2-2b, 6.2-2c, 6.3-8 and 6.3-9, Figure 5.4-9, Figure 5.4-11 and Figure 6.3-1. Additional mitigating features, such as use of reflective metal insulation (RMI) for large bore piping, Inservice Inspection Program as a Surveillance Requirement, temporary filters during post-construction system testing, and a foreign material exclusion program are introduced. Tables 6C-1 and 6C-2 have been deleted since they are not applicable to the new strainer design.

There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change.

This departure changes type of ECCS suction strainers and provides design methodology for these strainers. The cassette type strainer improves upon the conical type strainer in alleviating strainer blockage. The strainer design incorporates requirements outlined in latest Regulatory Guide 1.82 (Rev.3). These strainers are designed as follows.

(1) The head loss calculation was performed using head loss equation in NUREG/CR-6224. It was experimentally verified that this calculated head loss overestimates the actual head loss.

(2) The hydrodynamic and structural analysis calculations for submerged strainers can be performed on S/P penetrations, tee pipes and the strainers. These load calculation procedures take into consideration the effect of size, geometry, porosity and location of the new strainers.

Evaluation Summary

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, as described previously. The change affects the function of the strainers, but represents an improvement in safety and ensures that the design is bounded by the safety analysis.

The strainers are passive components that do not impact the operation of the ABWR and therefore do not impact the frequency of occurrence of an accident. Since the new strainers reduce the risk of blockage and improve the reliability of RHR, there is no adverse impact on the likelihood of malfunction of an SSC, the consequences of an accident, and the consequences of a malfunction of an SSC important to safety. Since the strainer improves the performance of the RHR system in an accident scenario, there is no impact on any fission product barriers. The new strainer does use a different methodology for evaluation, however, this methodology has been approved by the NRC in NUREG-6224. This change increases suction strainer surface area and reduces the risk of blockage and therefore there is no adverse impact on the severe accident evaluations. For this reason, prior NRC review and approval is not needed.

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

Description

The Technical Specifications (TS) for STP 3 & 4 include the allowable values in accordance with NUREG-1434, Rev. 3. This NUREG provides a detailed discussion on the specifics regarding the new Allowable Value (AV) single column format that it adopts. The purpose of this departure is to clarify in the FSAR that wherever the TS are referenced for setpoints or margins, the correct reference is to the methods for calculating the setpoints and margins as described in the TS Bases. Setpoints for high radiation levels are in accordance with the Offsite Dose Calculation Manual. Also, references to the TS are deleted if not necessary or if they need to be replaced with another proper reference.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. In summary, the departure clarifies the TS reference in the FSAR for setpoints and margins. The TS themselves are not being updated or reformatted under this departure. This departure does not change any Tier 1 or Tier 2* information, the Technical Specifications or Bases, or other operational requirements. Since the change involves only the reference substitution or deletion in the plant specific DCD (FSAR), there is no change to any SSC important to safety or fission product barrier, or any evaluated technical information. No change is made, due to this departure, to any method used for evaluation in establishing the design basis or in the safety analyses. Also, there is no effect on any accident evaluated previously, and no accident of a different type can occur. Since no design feature is changed, there is no impact on any method of mitigation of ex-vessel accident that relies on certain design features. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.1-2, ATWS DB for Startup Range Neutron Monitoring

Description

Subsection 7.1.2 of the reference ABWR DCD described the safe shutdown systems I&C and the Neutron Monitoring System I&C. The STP 3 & 4 FSAR provides the following departures:

- DCD Subsection 7.1.2.4.1 states that the FMCRD motors shall be connected to the emergency diesel generators. The FSAR subsection clarifies that power for the stepping motor driver modules (SMDMs) that control the power to the FMCRD motors derive their power from a bus that can automatically receive power from the EDG, if necessary.
- STP 3 & 4 FSAR Subsections 7.1.2.6.1.1 (1) and 7.1.2.6.1.4 (1) add as a General Functional Requirement under the Safety Design Bases that the Startup Range Neutron Monitoring (SRNM) subsystem and the Average Power Range Monitor (APRM) subsystem, respectively, will provide ATWS permissive signals to the ESF Logic and Control System (ELCS). These changes are also reflected in Table 7.6-5.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2* DCD, Technical Specifications, Bases for Technical Specifications or operational requirements.

The first item of the departure clarifies in more detail how the FMCRD motors receive power from the EDG. The second item reflects the existing design as described in Section 7.6.1.1.1(6) of the DCD whereby the SRNM sends an "ATWS Permissive" signal to the SSLC system to permit ATWS protection action, and the existing design in Section 7.6.1.1.2.2(5) of the DCD whereby the APRM sends an ATWS permissive signal to the SSLC system. These changes are for added clarification and completeness, do not affect the underlying design of any plant SSC important to safety, and do not impact the likelihood or consequences of an accident or malfunction of an SSC important to safety nor is there any impact on ex-vessel severe accident likelihood or consequences. No new accident scenarios are created by this change and no fission product barrier design basis limits are impacted. Therefore this change has no adverse impact and prior NRC approval is not required.

STD DEP 7.2-2, Description of Scram Actuating Relays

Description

DCD Subsection 7.2.1.1.4.1 (3) describes normally closed relay contacts in the scram logic circuitry between the air header dump valve solenoids (back-up scram solenoids) and the power source (125 VDC) for the air header dump valve solenoids.

The STP 3 & 4 FSAR subsection has revised the wording of the relay logic contact status from “normally closed” to “normally open” and clarified that the tripped state is when the coil is “energized.” This departure ensures a clear description is provided for the Reactor Protection System.

Evaluation Summary

This departure has been evaluated pursuant to and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This departure does not change the Technical Specifications, any underlying design or other operational requirements. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.2-4, Manual Scram Monitoring

Description

Subsection 7.2.1.1.4.2 (6) (c) of the reference ABWR DCD describes the two manual scram switches or the reactor mode switch as providing the means to manually initiate a reactor trip. The subsection also states that one bypass initiating variable is also monitored in addition to the scram initiating variables. This departure deletes the statement about monitoring initiating variables because it is misplaced in the Manual Scram subsection discussion and could be misleading in light of the manual scram not being bypassed.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change the Tier 1, Tier2*, Technical Specifications, bases for the Technical Specifications, nor any underlying design or other operational requirements. Furthermore, it does not change any plant physical features, SSCs important to safety, or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation, or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.2-6, RPS Instrumentation Ranges

Description

Table 7.2-1 of the reference ABWR DCD provided specifications for Reactor Protection System Instrumentation. This departure provides new ranges for:

- Reactor Vessel High Pressure
- Drywell High Pressure
- Reactor Vessel Low Water Level 3
- CRD charging header pressure High
- Turbine Control Valve Fast Closure
- High Suppression Pool Temperature
- Turbine First-stage Pressure

Continuing design effort has determined that the original ranges did not provide for optimal performance. The ranges are now updated to reflect a range of values appropriate for optimal performance.

Evaluation Summary

This departure to update the RPS instrumentation ranges has been evaluated pursuant to and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change the Technical Specifications, any underlying design or other operational requirements. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.3-1, Time Intervals for Licensing Analysis

Description

Subsections 7.3.1.1.1.1 and 7.3.1.1.1.4 of the reference ABWR DCD provide specific times for the High Pressure Core Flood System and the Low Pressure Flooder Subsystem to respond to accidents. Table 6.3-1 provides these same values in addition to other significant input variables used in the Loss-of-Coolant Accident analysis.

To ensure consistency of information within the DCD, the specific values have been deleted from these Chapter 7 subsections and a reference has been inserted to Table 6.3-1. This ensures that all data relative to these inputs remain in one place, consistent with the accident analysis.

Evaluation Summary

This departure has been evaluated pusuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

Specific response times for the Low Pressure Flooder and the High Pressure Core Flood systems exist both in the text narrative in Section 7.3 and in tables in Section 6.3 of the DCD. This change removes the specific times from the text narrative in Section 7.3 and references the appropriate table in Section 6.3. This removes the possibility of there being inconsistencies in the COLA. This change is intended to improve the quality and consistency of the information in the COLA and does not change any design or function for an SSC important to safety. Consequently, this change has no impact on the frequency or consequences of any accident or malfunction of an SSC important to safety previously evaluated. There is no impact on ex-vessel severe accident likelihood or consequences.

This change has no impact on any Tier 1, Tier 2*, Technical Specifications, bases for the Technical Specification, or operational requirements information.

Based on the results of this evaluation, prior NRC approval of this change is not required.

STD DEP 7.3-2, Automatic Depressurization System (ADS) Operator

Description

Subsection 7.3.1.1.1.2 of the reference ABWR DCD incompletely describes actuation of the automatic safety/relief valves as “with electrical power.” The valve utilizes pneumatic action for the relieving function, but the operating air is introduced via an electric signal to a solenoid valve. The relief (power) mode of operation is initiated when an electrical signal is received at any of the solenoid valves located on the pneumatic actuator assembly. These valves also operate by mechanical function as described in this subsection. The STP 3 & 4 FSAR states “pneumatic action” as the actuation method to clearly describe the ADS function of the SRV.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The change is a rewording for clarification that does not change the meaning or intent and has no impact on the ADS system design or function. As a result, there is no impact on the probability or the consequences of an accident or malfunction of an SSC important to safety. There is also no impact on the likelihood or the consequences of an ex-vessel severe accident. There is no impact on any Tier 1, Tier 2*, Technical Specifications, bases for Technical Specifications, or operational requirements information as a result of this change.

Based on the results of this evaluation, prior NRC approval of this change is not required.

STD DEP 7.3-4, ADS Logic

Description

Subsection 7.3.1.1.1.2 (3) (b) of the reference ABWR DCD describes the logic and sequencing for the ADS. The original description did not clearly describe the conditions under which ADS could be initiated. The description identifies the two parameters required as Reactor Water Level and Drywell Pressure. The description could be misinterpreted as requiring both parameters simultaneously to initiate ADS. The actual logic includes a bypass timer that initiates on Reactor Water Level (Level 1) that will initiate ADS without the presence of High Drywell Pressure after eight minutes (nominal). Subsection 7.3.2.1.1 discusses this timer, but does not provide the information that it is initiated by the Level 1 signal.

The above subsections are amended in the STP 3 & 4 FSAR to state that the bypass timer is initiated by the Reactor Vessel Water Level (Level 1) input. Additionally, the 8-minute exact value for the timer setting is removed from Subsection 7.3.2.1.1 to ensure that there is no conflict with Tier 1 information regarding the setting of less than or equal to 8 minutes for this timer.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change the Technical Specifications, setpoints for the parameters or other operational requirements. Furthermore, it does not change the ADS design, any plant physical features, or SSCs important to safety or fission product barriers. Previously evaluated accidents are not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. The ADS has not been identified as a design feature in the plant specific DCD for mitigating an ex-vessel severe accident. Consequently, this change has no impact on ex-vessel severe accident likelihood or consequences. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.3-5, Water Level Monitoring

Description

Subsection 7.3.1.1.1.2 and 7.3.1.1.1.4 of the reference ABWR DCD describe the equipment design for the ADS and RHR/LPFL I&C using the terms “Low” and “Low-Low” when describing the initiation inputs from the Reactor Water Level instrumentation. These terms are replaced by the standard ABWR nomenclature of Level 1.5 and Level 1, respectively, for initiating signals. This instrumentation also provides initiating signals for other levels, such as Level 2, Level 3, and Level 8, etc.

To ensure clarity for all users, terms such as “Low” and “Low-Low” are replaced with the actual level nomenclature, e.g., “Level 1.5” and “Level 1,” in Subsections 7.3.1.1.1.2 (3) (a), (3) (b) and (3) (d) and 7.3.1.1.1.4 (3) (a), (3) (d), and (3) (e).

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5and determined not to require prior NRC approval. There is no underlying design change to any plant SSC. In summary, the evaluation demonstrates that this departure is a clarification since it provides the standard ABWR nomenclature for the reactor vessel level initiating instrumentation. This departure does not change any Tier 1 or Tier 2* information, change the Technical Specifications, or other operational requirements, and does not affect any previously evaluated accident or create the possibility for an accident of a different type. Further, there is no effect on any SSC important to safety or fission product barrier, and it does not affect any method used for evaluation in establishing the design basis or in the safety analyses. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.3-6, SRV Position Indication

Description

Subsection 7.3.1.1.1.2 (3) (b) of the reference ABWR DCD describes the position indication provided in the main control room for the safety/relief valves. The description states that lights are provided when the solenoid-operated pilot valves are energized to open. It also states that linear variable differential transformers (LVDTs) are mounted on the valve operators.

As stated in the STP 3 & 4 FSAR subsection, the current design for main control room indication of safety/relief valve position is provided by a limit switch. ADS solenoid energized status is not indicated as this is not a direct indication of the safety/ relief valve position. The incorporation of the limit switch on the valve provides a direct, positive indication of the safety/relief valve position that is more reliable than the original described LVDT. The requirement for position indication of the safety/relief valve is assured by this limit switch.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change.

This change provides direct position indication of the safety relief valves using limit switches in place of the method described in the DCD where position of the safety relief valves was monitored with less reliable LVDTs or inferred by indication of energization of the solenoids to the pilot valves. The direct monitoring of SRV position with the more reliable limit switches versus LVDTs provides a more reliable and direct indication of

SRV position status to the control room operator. There are no accidents described in the DCD or the COLA that are directly affected by this change, thus there is no adverse effect on the frequency or consequences of accidents, and the change does not affect the occurrence or consequences of a malfunction of SSCs important to safety. No accident analyses of fission product barrier design limits are impacted by this change. The change to SRV position indication swtiches has no impact on frequency of occurrence or consequences of ex-vessel severe accidents previously reviewed.

The result of the evaluation is that prior NRC approval is not required for this change.

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

Description

Subsection 7.3.1.1.1.2(3)(b) of the reference ABWR DCD describes the manual controls associated with the ADS. This section describes the ADS inhibit switch as “keylocked.” The ADS inhibit switches are no longer the keylock type. The subsection is modified in the STP 3 & 4 FSAR to present the current design.

Evaluation Summary

DCD Tier 1 and Tier 2* information, Technical Specifications, Technical Specification Bases, and operational requirements were reviewed and were found to not be impacted by this change.

This departure replaces the ADS inhibit switches from a keylock type to a normal manual switch. This change does not affect the overall function of the ADS inhibit switches. This change is to facilitate operator action.

The purpose of the ADS inhibit switch is to allow one ADS division to be taken out of service. This switch is ineffective once the ADS timers have timed out and thus cannot be used to abort and reclose the valves once they are signalled to open. The inhibit mode is continuously annunciated in the main control room. This departure only changes the type of switch and does not change the functionality of ADS.

This departure does not change any of the automatic initiation capabilities of the ADS on such signals as low reactor water level and high drywell pressure. Because there is no change in design or function of an SSC important to safety. This departure has no impact on the likelihood or consequences of analyzed accidents or of a malfunction of an SSC important to safety. No new accident scenarios are created and there is no impact on the design basis limit of any fission product barrier. The ADS has not been identified as a design feature in the plant specific DCD for mitigating an ex-vessel severe accident. Therefore this change has no impact on the likelihood or consequences of an ex-vessel severe accident.

Based on the results of this evaluation, prior NRC approval of this change is not required.

STD DEP 7.3-9, Shutdown Cooling Operation

Description

Subsection 7.3.1.1.1.4 (3) (e) of the STP 3 & 4 FSAR clarifies the reference ABWR DCD description of the RHR Shutdown Cooling Mode valve alignment after a Low Pressure Flooder (LPFL) actuation signal. In the shutdown cooling mode of operation, the RHR System removes decay heat from the reactor core to achieve and maintain a cold shutdown condition. In this mode, each division takes suction from the RPV via its dedicated suction line, pumps the water through its respective heat exchanger tubes, and returns the cooled water to the RPV. Each shutdown cooling suction valve automatically closes if reactor water level falls below Level 3. These valves will not open on high reactor pressure.

This clarification deletes reference to automatic closure of the RHR suction valves for the SCS mode on receipt of an LPFL initiation signal on Level 1. These valves are already automatically closed on a Level 3 signal. This departure further clarifies that the shutdown cooling isolation valves must be closed to permit suction from the Suppression Pool.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1 or Tier 2* information, the Technical Specifications, any underlying design or other operational requirements. This departure clarifies the correct alignment of RHR valves in the SCS mode upon receipt of an LPCF initiation signal and does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation or an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.3-10, ESF Logic and Control System (ELCS) Mode Automation

Description

Subsection 7.3.1.1.1.4(3i) of the reference ABWR DCD states that the operator may control the RHR pumps and injection valves manually after LPFL initiation by using RHR capabilities in other modes if the core is being cooled by other emergency core cooling systems. Subsection 7.3.1.1.1.4(3i) of STP 3 & 4 COLA replaces that statement with an expanded description of the Mode switches in the main control room. In order to support the displays and to reduce operator burden, RHR has specific mode operation capability. This eliminates the possibility of operator error and supports the display requirements. Mode-specific permissives are required for system alignment.

The change to Section 7.3.1.1.1.4 replaces the DCD text which states that the operator may control the RHR pumps and injection valves manually after LPFL initiation to use RHR capabilities in other modes if the core is being cooled by other emergency core cooling systems. The updated text expands this by describing the Mode switches in the main control room. These logic changes are reflected in a revision made to Figure 7.3-4, Sheets 10, 11, 13, 14, 19, 20 and 20a.

In addition, Section 7.3.1.1.1.4 (3) explains that logic power for the LPFL subsystem is from the ELCS power supply for the division involved. ELCS mode automatic logic changes are implemented for Figure 7.3-1, Sheets 5, 11, 13, 14 and 17 to assure that the HPCF “C” diverse hard-wired manual initiation function has priority over the normal automatic initiation logic for HPCF “C”. This departure also modifies information in Figure 7.3-1, Sheets 2, 7-10, 15 and 16. These changes assure proper implementation of the diverse hard-wired HPCF “C” manual initiation capability described in Appendix 7C.5.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

The Tier 1 and Tier 2* DCD, Technical Specifications, Bases for Technical Specifications and operational requirements were reviewed and it was found that none of these documents are impacted by this change.

The change to Section 7.3.1.1.1.4 replaces the DCD text which states that the operator may control the RHR pumps and injection valves manually after LPFL initiation to use RHR capabilities in other modes if the core is being cooled by other emergency core cooling systems. The updated text expands this by describing the Mode switches in the main control room. In order to support the displays and to reduce operator burden, RHR has specific mode operation capability. These changes are also reflected in revisions made to Figure 7.3-4. These changes reduce the likelihood of operator error and support the display requirements. The logic changes to the ELCS are implemented to assure that the HPCF "C" diverse hard-wired manual initiation function has priority over the normal automatic initiation logic for HPCF "C". These changes assure proper implementation of the diverse hard-wired HPCF "C" manual initiation capability described in Appendix 7C.5.

These changes provide enhancements to reduce the likelihood of operator error, to support display requirements and to implement manual diversity as described in Appendix 7C.5. Consequently, these changes are favorable and have no adverse impact on the frequency or consequences of an accident or malfunction of an SSC important to safety. The changes do not create an accident of a different type than previously evaluated and do not adversely affect a fission product barrier limit. These changes also have no adverse effect on the likelihood or consequences of an ex-vessel severe accident.

Based on the results of this evaluation, prior NRC approval of this change is not required.

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

Description

Reference ABWR DCD Subsection 7.3.1.1.2 (3)(l) provides a description of the leak detection instrumentation provided for valve stem leak-off lines of large bore reactor coolant pressure boundary isolation valves. Originally, valve stem packing rings were mostly made of asbestos material, which was prone to shrinkage during service. The shrinkage could cause voids in the packing chamber, which leads to leakage. To counter frequent leaks, two sets of packing rings were provided with a leak-off line from the chamber between the packing rings. The leak-off was then routed to a collection sump, where leakage was identified in accordance with pressure boundary leakage requirements. While providing relief from leakage requirements, this arrangement did not resolve the issue of stem leakage.

To resolve the stem leakage issue, valves were specified in the FSAR to use one set of expanded graphite packing to seal the valve stem penetration. Expanded graphite has shown superior sealing properties, is less likely to induce corrosion and damage to the valve stem due to trace material, retain their form longer and avoid the formation of voids that could lead to leakage. Due to the valve packing changes, during the design evolution of the ABWR, the valve stem leak-off lines have been eliminated. The valve gland leak-off lines have been eliminated for the valves and the described instrumentation is no longer applicable. The large remote power operated valves, located in the Drywell for Main Steam, Reactor Water Clean Up, Reactor Core Isolation Cooling and Residual Heat Removal Systems are affected by this change. A similar discussion is provided in STP 3 & 4 FSAR Subsection 5.2.5. The deletion of this section ensures the discussion of the Leakage Detection Instrumentation is consistent with the current design.

Evaluation Summary

This departure to eliminate the RCS isolation valve stem/gland leakage monitoring system has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This change is removal of the RCS isolation valve leak-off lines piping arrangement and the instrumentation for valve stem leakage monitoring.

This departure does not change any Tier 1, Tier 2* information, the Technical Specifications, bases for technical Specifications, any underlying design or other operational requirements.

The improvements in valve packing have changed the way pressure boundary leakage from valves is assessed. The packing improvements have reduced valve stem leakage and result in a more reliable configuration. With the reduced stem leakage, and more reliable configuration, a leakage detection system is no longer needed. This conclusion results in the removal of the piping arrangement and the instrumentation for direct monitoring of valve stem leakage detection.

It does not change any functional or safety requirements to monitor and assess valve leakage. Existing systems for detection of unidentified leakage (e.g. drywell floor drain sump monitoring) as described in Subsection 5.2.5.2.1 of the STP 3 & 4 COLA are capable of monitoring and alarming any leakage from these valves, including valve stem leakage.

Since this change does not affect any plant SSC except for removing the valve leak-off piping and associated monitoring instrumentation, there is no effect on any accident previously evaluated in the DCD. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.3-13, Containment Spray Logic

Description

The reference ABWR DCD states that if Containment Spray has been initiated, then the system automatically realigns to the LPFL Mode if Reactor Vessel Water Level falls below Level 1. This departure changes STP 3 & 4 FSAR Subsections 7.3.1.1.3 (3) (a), (b), and (c), and 7.3.2.3.2(1) and (4) to; 1) emphasize that the LPFL mode has precedence over containment Spray when below Level 1, 2) clarify the method by which the Drywell and Wetwell sprays can be initiated, and 3) clarify the interlocks associated with this mode of RHR operation. Figure 7.3-4 sheets 4, 6, 10, 11, 13 & 20 are revised to reflect logic changes for removal of the manual override logic for the wetwell spray valves and suppression pool return valves. The annunciator status lights for these functions are removed from the table of status lights and annunciators.

Evaluation Summary

This departure clarifies the operation of the Containment Spray System, removes the manual override logic, and provides a more complete description of the operation of this mode of RHR. This departure has been evaluated in accordance with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The containment spray system is discussed in Tier 1 and Chapter 15. Those sections have been reviewed and the departure has no adverse effect on them. The design change to remove the manual override logic is made to reflect that the spray system will continue to operate until manually terminated by the operator, or will automatically terminate and realign to the LPFL injection mode on receipt of a RPV Water Level 1 since core cooling has priority. There is no adverse effect on any SSC important to safety or fission product barrier. There is no increase in the frequency of accidents and no increase in the likelihood of a malfunction of an SSC important to safety. Any previously evaluated accident is not affected and the possibility for an accident of a different type is not created. Also, it does affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation to an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.3-14, Residual Heat Removal Suppression Pool Cooling Logic

Description

Subsection 7.3.1.1.4 (3) (b) of the reference ABWR DCD describes the logic and sequencing of the RHR Suppression Pool Cooling Mode. The FSAR includes the following departures:

- Item (ii) has been corrected to show that valves in other RHR modes are automatically repositioned to align to the SPC mode.
- A description of the Suppression Pool Cooling Mode Switch has been added to Item (iv) to provide a more complete understanding of the initiation of this mode.
- Item (v) of this subsection has been augmented to state the SPC mode continues to operate until the operator closes the discharge valves. This operation is facilitated by the activation of another permitted mode of operation. This information has been added to ensure a complete understanding of the termination of this mode.
- Item (vi) has been added to this section to clarify that this mode only operates automatically when entered from the RHR Standby Mode.
- Figure 7.3-4 has been revised to change the SPC manual initiation switch from an “On/Off” to an “Arm/Disarm” type.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This change has no impact on Tier 1, Tier 2*, technical specifications, basis for technical specifications and operational requirements. The logic and sequencing of the automatic and manual actions required to initiate and terminate the operation of the suppression pool cooling mode of the RHR system are correctly defined in ABWR DCD Tier 2 subsection 5.4.7.1.1.5. The changes to Section 7.3.1.1.4 provide a more complete description of the SPC mode automatic and manual operations and makes them consistent with the description in Section 5.4.7.1.1.5. The design change to an Arm and Initiate switch provides added assurance that the operator will not inadvertently switch to the SPC mode from the LPFL mode while performing the critical operation of maintaining water level in the RPV. This change does not impact the automatic initiation of the SPC mode on high SP temperature. Consequently, this change will not impact the frequency or consequences of accidents or malfunction of an SSC important to safety. There is no impact on any fission product barrier, nor is there any impact on the likelihood or consequences of an ex-vessel severe accident.

Consequently, prior NRC approval of this change is not required.

STD DEP 7.3-15, Reactor Service Water Logic Interfaces

Description

Subsection 7.3.1.1.7(3i) of the reference ABWR DCD provides information about the safety interfaces for the Reactor Cooling Water controls. This description is modified in the FSAR Subsection as follows:

- The original information stated that only Division I and II provided flow signals to the Main Control Rooms. The current design provides flow signals for all three divisions (Div. I, II, and III) of RCW.
- This section also discusses the “RCW Hx A or D” differential pressure instrumentation. This equipment is actually the strainers on the discharge side of the two RSW pumps in each division. Therefore, the nomenclature of this equipment is changed to “RSW A or D strainer.”

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

The Tier 1 and Tier 2* DCD, technical specifications, basis for technical specifications and operational requirements were reviewed and were found to not be impacted by this change.

The first bullet item in the departure description is an improvement as it provides the control room operator with additional and more complete information regarding RCW flow. The second bullet item changes the annunciator alarm on high differential pressure from the RCW heat exchangers to the RSW A or D strainers. It does not change any control room indications concerning these systems (e.g. flow rate, differential pressure, temperature) nor does it change any SSC important to safety. Because fouling of the strainers is more likely to restrict cooling flow in the RSW than fouling of the heat exchangers, providing differential pressure monitoring of the strainers will provide more effective monitoring of conditions that could impede flow in the RSW. Based on this discussion, this change will not have an adverse affect on the reliability of heat removal capability for the RCW/RSW. Consequently, there is no adverse impact on the likelihood or consequences of an accident or malfunction of an SSC important to safety previously evaluated. There are no new accident scenarios created as a result of this change nor is there any impact on a fission product barrier design basis limit. The likelihood or consequences of an ex-vessel severe accident are not impacted by this change.

Based on this evaluation, prior NRC approval of this change is not required.

STD DEP 7.3-16, Testing Safety Relief Valve Solenoid Valves

Description

Improved testing capabilities have been incorporated into the ABWR design compared with those described in of Subsection 7.3.1.1.1.2 (g) of the reference ABWR DCD. These improvements allow the testing of the safety/relief valve pilot solenoid valves to be performed at any pressure. Therefore, the restrictions that were discussed in the reference ABWR DCD are no longer applicable and have been removed.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

The Tier 1 and Tier 2* DCD, Technical Specifications, Basis for Technical Specifications and operational requirements were reviewed and were found to not be impacted by this change.

Improved testing capabilities have been incorporated into the ABWR design described in Subsection 7.3.1.1.1.2 (g) of the reference ABWR DCD. These improvements allow the testing of the safety/relief valve pilot solenoid valves to be performed at any pressure instead of only when the reactor is not pressurized. By allowing testing at any pressure, flexibility for testing of these valves is enhanced. Consequently, this change has no adverse affect on the frequency or consequences of accidents or malfunction of an SSC important to safety. There is no adverse impact on any fission product barrier design basis limits, and there is no failure of an SSC leading to a different result than previously analyzed. The ADS has not been identified as a design feature in the plant specific DCD for mitigating an ex-vessel severe accident. Consequently, this change has no impact on ex-vessel severe accident likelihood or consequences.

As a result of this evaluation, prior NRC review of this change is not required.

STD DEP 7.4-1, Alternate Rod Insertion (ARI) Function Description

Description

This departure revises Subsections 7.1.1, 7.1.2, 7.4.1 and 7.4.2 of the STP 3 & 4 FSAR . The original description in the reference ABWR DCD described the implementation of the ARI function for the following features. The revised STP 3 & 4 FSAR wording clarifies the descriptions of these features. It specifies the:

- ARI function is a diverse method for inserting control rods by either hydraulic insertion or Fine Motion Control Rod Drive (FMCRD) motor run-in by providing a more complete discussion of the function.
- Low-level signals or high vessel pressure from the safety systems (i.e., SSLC-ESF) and the RFC system for the ARI function are isolated. The interface of the isolated signals from safety system to the non-safety RFC system ensures that no safety related functions are affected.
- Two dedicated switches on the Main Control Room Panel located near the RCIS dedicated operator interface to clarify the manual initiation capability.
- Complete scope of the key components related to the ARI function.

This departure provides a clear and concise understanding of the ARI function, which is not required for safety, nor are its components considered Class 1E. This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

Evaluation Summary

The Tier 1 and Tier 2*, technical specifications, basis for technical specifications and operational requirements were reviewed and were found to not be impacted by this change.

This departure does not change the design nor the functioning of the ARI. It’s purpose is to provide additional details on the functioning of the instrumentation and controls for this nonsafety-related system. The ARI system is described in several sections of the DCD such as 7.1.1.4.1, 4.6.1.2.5.4 and 19.3. It is described as a nonsafety-related system which is separate and diverse from the RPS safety-related shutdown system, and initiated by low reactor water level or high vessel pressure signals or by manual means. This departure does not change these basic characteristics of the system but only adds further details on each of these items. Because there is no change in design or function of an SSC important to safety, this departure has no impact on the likelihood or consequences of analyzed accidents or of a malfunction of an SSC important to safety. No new accident scenarios are created and there is no impact on the design basis limit of any fission product barrier. The ARI has not been identified as a design feature in the DCD for mitigating an ex-vessel severe accident, and therefore this change has no impact on the likelihood or consequences of and ex-vessel severe accident.

Based on the results of this evaluation, prior NRC approval of this change is not required.

STD DEP 7.4-2, Residual Heat Removal (RHR) Alarm

Description

Subsection 7.4.2.3.1 of the reference ABWR DCD provides functional requirements for the reactor shutdown cooling mode of the RHR system. Item (3) of this section provides a list of alarms that apply to all modes of the RHR System. As a result of detailed design evolution, the STP 3 & 4 FSAR replaces the alarm for “RHR Logic Power Failure” with the more general alarm “ELCS Out of Service.” The FSAR also clarifies that the only time the “Manual Initiation Armed” alarm is activated is when the RHR system is in the Low Pressure Flooder (LPFL) Mode of operation.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

The Tier 1 and Tier 2* DCD, Technical Specifications, Bases for Techical Specifications and operational requirements were reviewed and were found to not be impacted by this change.

The replacement of the alarm for “RHR Logic Power Failure” with the alarm “ELCS Out of Service” reflects the STP 3 & 4 logic design whereby the logic for all of the ECCS (e.g. HPCF, ADS, LDS, RHR systems) is controlled and powered by the ELCS. Because the ELCS powers the logic for the RHR system, the “ELCS Out of Service” alarm description provides the control room operator with the same information as for the previous alarm description and therefore would have no effect on the information available to the operator to take actions following any accident or malfunction of an SSC.

The second change is a clarification as only the LPFL mode of the RHR system has an arming feature. Typically, when a switch is “armed” an alarm is activated alerting the operator that they are about to operate a system.

These changes do not impact any SCC design or function. The likelihood and the consequences of an accident or malfunction of an SSC important to safety are not impacted. There is no impact on a design basis limit for a fission product barrier nor is there any change to a method of analysis previously approved. There is no impact on the probability or consequences of an ex-vessel severe accident.

Based on the evaluation, prior NRC approval of this change is not required.

STD DEP 7.6-1, Oscillation Power Range Monitor (OPRM) Logic

Description

Subsection 7.6.1.1.2.2 of the STP 3 & 4 FSAR has been changed and clarified to now ~~states the~~state

- OPRM trip logic and input to RPS is performed separately from the APRM trip logic in RPS
- ~~Each~~The OPRM ~~channel~~function has its own inoperative trip~~, which actuates~~ when the channel has less than the required minimum operable cells, ~~a fault is detected~~or there is an OPRM self-test fault, the OPRM instrument watchdog timer has timed out, or if there is a loss of power to the ~~PRNM instrument~~OPRM.
- The trip signals from each division of APRM and OPRM are provided separately to the RPS, where the RPS performs two-out-of-four voting.
- The period tolerance factor (t_{error}) is limited between 0.100 and 0.300 seconds and the time constant of the averaging flux filter is set at 0.95 seconds.

Consistent with the changes above, Subsections 7.6.2.1.1 and 7.6.2.1.2 are also revised to reflect that the the OPRM is independent from the APRM.

At the Tier 1 level of design requirements, there is a brief presentation of the Neutron Monitoring System, which combines the system nomenclature and terminology with the safety and non-safety functions that are to be performed. Such presentation does not prevent implementation of robust separation of the functions to achieve enhanced reliability, fault tolerance, self-testing flexibility, and repairability of the different functions. The following information identifies the pertinent Tier 1 and Tier 2 information, which is interpreted to allow a robust design of the OPRM logic function separate from the APRM logic function, including separate trip inputs to RPS.

DCD Tier 1, Subsection 2.2.5, Neutron Monitoring System, states that the Oscillation Power Range Monitor (OPRM) is part of the APRM. Additionally, DCD Tier 1, Figure 2.2.5, Neutron Monitoring System, represents a NMS division configuration showing LPRM/APRM (includes OPRM) as an I/O function. DCD Tier 1, Table 2.2.5, Neutron Monitoring System ITAAC, item 2, states the design commitment as the OPRM protection provides trip output to the RPS. Item 6 states the design commitment as the APRM can generate high neutron flux trip, a STP trip signal, a rapid core flow decrease trip signal, or a core power oscillation trip signal. This is shown on COLA Tier 2, Figure 7.6-2, sheet 9.

In addition, the DCD Tech Specs 3.3.1.1 discuss the APRM and OPRM independently. Also, COL Item 7.2 implements the BWROG stability option III as evaluated by the NRC in NUREG-1503, Final Safety Evaluation Report Related to the Certification of the ABWR (FSER).

The revised COLA Tier 2 Figure 7.6-2, Neutron Monitoring System IBD, sheet 9b, shows that the OPRM logic is independent of the ARPM logic, but designated as part of the APRM channels. Revised Figure 7.6-2, sheet 27, also shows that the ARPM bypass also bypasses OPRM.

Consistent with the information above, Subsections 7.6.2.1.1 and 7.6.2.1.2 are also revised to clarify the independence of the OPRM logic from the APRM logic, and that the departure is consistent with Tier 1 information.

Evaluation Summary

~~This~~As described above, this design change to separate the OPRM trip logic and input to RPS from the APRM trip logic in RPS is an upgrade expected to reduce the likelihood of reactor scrams from separate failures of an OPRM and an APRM ~~channels~~channel without ~~impacting~~affecting performance of their separate safety functions. With this logic design, a trip in one APRM channel and in one OPRM channel does not result in a reactor scram. Any two OPRM channels that sense an abnormal condition will result in a reactor scram through the RPS. The BWR Owner's Group has endorsed this separate OPRM and APRM logic configuration and there has been initial favorable response from the NRC as exhibited in NUREG-1503 and COL Item 7.2. This logic configuration also is consistent with the trip logic design philosophy implemented in the remainder of the RPS.

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This departure does not change any Tier 1 or Tier 2* information, the Technical Specifications or Bases, any underlying design or other operational requirements. The change to separate the OPRM and APRM trip logic still meets the intent of the information in Tier 1, Subsection 2.2.5. Furthermore, it does not change any other plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created, nor is the consequence of any accident increased. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation or the consequences of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.6-2, SPTM Subsystem of Reactor Trip and Isolation System

Description

The reference ABWR DCD description for the Suppression Pool Temperature Monitoring (SPTM) System in Subsection 7.6.1.7.1 has been clarified in the STP 3 & 4 FSAR to add that SPTM System is a subsystem of the Reactor Trip and Isolation System (RTIS).

Evaluation Summary

This departure has been evaluated in accordance with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. Thisdeparture is clerical in that it adds further clarification as the result of STD DEP T1 3.4-1. Part 3 of STD DEP T1 3.4-1 in part states, “to better define the functional design and implementation of the digital controls platforms, specific I&C system names were assigned to the ESF digital controls systems and the Reactor Protection System (RPS).” It further states, “The RPS functions are implemented in two separate I&C systems: the Reactor Trip & Isolation System (RTIS) and the Neutron Monitoring System (NMS).”

Tier 1 changes have been reviewed separately. The SPTM System is not impacted by adding this clarification. This proposed change has no impact on Tier 1, Tier 2*, Tech Specs, bases for Tech Specs or operational requirements.

No underlying design change is made and no SSC important to safety or fission product barrier is affected. There is no increase in the frequency of accidents and there is no increase in the likelihood of a malfunction of an SSC important to safety. Any previously evaluated accident is not affected and the possibility for an accident of a different type is not created. Also, it does affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.Therefore, this change has no adverse impact and does not require prior NCR approval.

STD DEP 7.6-3, SPTM Sensor Arrangement

Description

The reference ABWR DCD description for the SPTM System in Section 7.6.1.7.3 is clarified in the FSAR to better illustrate the approximate temperature sensor locations in relation to the SRVs. The STP 3 & 4 FSAR rewording states that the SRV discharge line quenchers are in direct sight of two sets of SPTM system temperature sensors.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

The Tier 1 and Tier 2* DCD, technical specifications, basis for technical specifications and operational requirements were reviewed and were found to not be impacted by this change.

This departure provides additional clarification and detail regarding the location of the temperature sensors in the suppression pool. The DCD indicates that the temperature sensors are in direct sight of the SRVs. The SRVs are located in the drywell and are mounted on the main steam piping whereas the suppression pool monitors are located in the suppression pool in the wetwell. The intent of this statement in the DCD was to note that the discharge from the SRVs is located in proximity to the suppression pool temperature monitors in order to provide an early and reliable indication of suppression pool temperature rise as a result of a transient or accident condition. Consequently, it has been clarified that the temperature monitor location is with respect to the SRV line quenchers at the discharge into the suppression pool. This is a clarifying change only and does not impact any SSC design or function. As a result, there is no impact on the likelihood or consequences of analyzed accidents or failure of an SSC important to safety. There is no impact on a fission product design basis limit nor are there any new accident scenarios created by this change, as it is only a clarifying change. The SPTM system has been identified as a design feature in the DCD for mitigating an ex-vessel severe accident. However, because this is a clarifying change only and does not impact the system design or function, there is no increase in the probability or consequences of an ex-vessel severe accident.

Based on this evaluation, prior NRC approval of this change is not required.

STD DEP 7.6-4, Range of Power Range Neutron Monitoring Operability

Description

The reference ABWR DCD description for the Power Range Neutron Monitors (PRNM) in Subsection 7.6.2.1.1 stated that the PRNM provide information for monitoring the average power level of the reactor core and for monitoring the local power level when the reactor power is in the power range (above approximately 15% power). The FSAR clarifies the statement to indicate that the power range begins at approximately 5% power.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1 or Tier 2* information, Technical Specifications, bases for Technical Specifications, any underlying design or other operational requirements. This change corrects the bottom of the power range for operation of the PRNM from 15% to its actual value of 5%. Consequently, this change is favorable and correctly reflects the actual design, which provides overlap with the SRNM for neutron flux monitoring in the range of 5%-15% power. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not adversely affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-1, RPV Water Level Instrumentation

Description

Subsection 7.7.1.1 of the reference ABWR DCD implies that all instrument lines are flushed even when they do not need to be. A clarification indicates that only those instrument lines with a condensing chamber can have entrained non-condensible gasses. STP 3 & 4 FSAR Subsections 4.6.1.2, 7.7.1.1, 15B.2.3 and Figure 15B-1 now state that the concern of non-condensible gas build-up in the water column in the reactor vessel reference leg water level instrument lines, i.e. the reactor vessel instrument lines at the elevation near the main steam line nozzles, has been addressed by continually flushing these instrument lines with water supplied by the Control Rod Drive (CRD) System for those instrument lines with a condensing chamber.

The original design intent was to have flushing only apply to lines with condensing chambers which was not clear in the original DCD. Subsection 7.7.1.1 of the FSAR provides this clarification.

In addition, this departure also updates Section 4.6.1.2 and 15B.2.3 of the reference ABWR DCD to clarify that the Control Rod Drive Hydraulic System (CRDHS) supplies the purge flow for the NBS instrument lines.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This departure does not change any Tier 1, Tier 2* information, the Technical Specifications or Bases, any underlying design or other operational requirements.

This departure clarifies that the source of water for purging of the instrument lines in the Nuclear Boiler System (NBS) is the CRD hydraulic system (CRDHS). It further clarifies that for the RPV level instruments, this purging is only performed on instrument lines with condensing chambers. The purpose of the instrument line purging is to eliminate any non-condensible gases which could lead to erroneous level indications. Because only those RPV level instruments with condensing chambers have the potential for buildup of non-condensible gases, this purging is only required for those lines. The function of the purging system for removal of non-condensible gases from instrument lines is unaffected by this change. This departure has no impact on any SSC system design intent or function and has no impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-2, SRV Discharge Pipe Temperature Data Recording

Description

There have been significant technological advances in data recording since the reference ABWR DCD was written. Subsections 7.3.1.1.1 and 7.7.1.1 of the FSAR now state that the discharge temperatures of all the safety/relief valves are shown on an historian function in the control room.

Recording SRV discharge temperature data is now performed in a more accurate manner and is easily retrievable. The recorded data rate meets all design criteria. The data recorded remains the same along with the parameters.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1 or Tier 2* information, the Technical Specifications, the bases for Technical Specifications, any other underlying design or other operational requirements.

SRV discharge pipe temperature data recording and alarm change is a design upgrade to replace the multipoint recorders with a historian function digital system. It does not adversely affect any functional or safety requirements for temperature monitoring, does not affect temperature detection or high temperature alarm setpoints, and the data recorded and parameters remain the same. Since this change does not affect any other plant SSCs, there is no effect on any accident previously evaluated in the DCD. Furthermore, it does not change any plant physical features other than that affected by this design change, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.Therefore this change has no adverse impact and prior NRC approval is not required.

STD DEP 7.7-3, Feedwater Turbidity

Description

The reference ABWR DCD Subsection 7.7.1 discusses the measurement of feedwater turbidity and signal transmission to the MCR, but turbidity is normally determined by sampling. There is no practical manner in which to perform this measurement. Because of this, and since measurement of turbidity is not considered to have any safety significance, it is being deleted.

Evaluation Summary

This departure to remove the feedwater turbidity monitoring subsystem has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1 or Tier 2* information, the Technical Specifications and Bases, any underlying design or other operational requirements.

It does not change the functional or safety requirements of the feedwater or condensate systems. Filtering of corrosion products and other impurities from the feedwater system is accomplished by the condensate purification system and the reactor water cleanup system. The adequacy of performance of these systems is indicated by instruments such as conductivity monitors to assure adequate purity of water flowing to the reactor vessel. Any determination of turbidity for a system such as feedwater, if needed, could be performed through sampling techniques. Furthermore, this change does not impact any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-4, Automatic Power Regulator / Rod Control and Information System Interface

Description

Section 7.7.1.2 (1) (a) (ii) of the reference ABWR DCD described the Power Generation and Control System (PGCS) as initiating control changes in the automatic rod movement mode. The STP 3 & 4 FSAR now clarifies that the APR is actually the direct controlling system that interfaces with the RCIS for accomplishing automatic rod movement mode and the PGCS interfaces only with APR for initiating various reactor power change control tasks.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, .

The Tier 1, and Tier 2*, DCD, Technical Specifications, basis for Technical Specifications and operational requirements were reviewed and not impacted by this change.

This departure clarifies and corrects Section 7.7.1.2(1)(a)(ii) of the DCD. As noted in Section 7.7.1.5.2 of the DCD, “The PGCS contains the algorithms for the automated control sequences associated with plant startup, shutdown and normal power range operation. The PGCS issues reactor command signals to the automatic power regulator (APR). The reactor power change algorithms are implemented in the APR." Section 7.7.1.2 was updated to be consistent with this DCD description which reflects the proper role of the APR and PGCS systems. This departure is a clarification only and does not affect the design or function of any SSC important to safety. As a result, there is no adverse impact on the likelihood or consequences of accidents or malfunction of any SSC important to safety. There are no new accident scenarios created as a result of this change nor is there any change to a fission product barrier design basis. The likelihood and consequences of ex-vessel severe accidents are not impacted by this change.

As a result of this evaluation, prior NRC review of this change is not required.

STD DEP 7.7-5, Rod Control and Information System (RCIS) Display

Description

Subsection 7.7.1.2 (1) (b) of the STP 3 & 4 FSAR clarifies the wording of the reference ABWR DCD by providing more precise information about available display information at the RCIS dedicated operator interface on the main control panel.

Evaluation Summary

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The change has no adverse impact.

The Tier 1 and Tier 2* DCD, Technical Specifications, basis for Technical Specifications, and operational requirements were reviewed and are not impacted by this departure.

This departure clarifies descriptions to provide more complete design descriptions. This departure dose not affect the design or function of any SSC important to safety. As a result, there is no adverse impact on the likelihood or consequences of accidents or malfunction of any SSC important to safety. There are no new accident scenarios created as a result of this change nor is there any change to a fission product barrier design basis. The likelihood and consequences of ex-vessel severe accidents are not impacted by this change.

As a result of this evaluation, prior NRC review of this change is not required.

STD DEP 7.7-6, Rod Control and Information System Commands

Description

ABWR DCD Tier 2 Subsection 7.7.1.2 (1) (f) describes the command signal from the Recirculation Flow Control System (RFCS) to the Rod Control and Information System (RCIS) for the Alternate Rod Insertion (ARI) and Subsection 7.7.1.2 (1) (g) describes the command signal from the RFCS to the selected control rod run-in (SCRRI). This COLA change implements the following clarifications to Subsections 7.7.1.2 (f) and (g):

- Subsection 7.7.1.2 (1) (f) is revised to clarify that redundant command signals (more than a single signal) are sent from RFCS to RCIS for the ARI function.
- Subsection 7.7.1.2 (1) (g) is revised to clarify that redundant command signals (more than a single signal) are sent from RFCS to RCIS for the SCRRI function.

Evaluation Summary

These changes are consistent with the details of RCIS IED (Figure 7.7-2) and with the description of the command signals as provided in Section 7.7.1.2.2 (2) of the DCD.

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. Therefore this change has no adverse impact and prior NRC approval is not required.

This departure does not change any Tier 1, Tier 2* information, the Technical Specifications or Bases, any underlying design or other operational requirements.

This departure is a clarification to Sections 7.7.1.2 (1) (f) and 7.7.1.2 (1) (g) of the DCD to make them consistent with the description of the command signals to the RCIS from the RFCS for the ARI and SCCRI functions, respectively, as provided in more detail in Section 7.7.1.2.2 (2) (a) and (b) of the DCD. In those sections, it is noted that the three channels of the RFCS provide each of the two channels of the RCIS logic with the ARI and SCRRI signals. RCIS internal logic to initiate the RCIS ARI and SCRRI functions is based on two-out-of-three logic within each channel of the RCIS. Consequently, initiation of the ARI and SCRRI functions is based on multiple signals from the RFCS.

The description of that initiation was accordingly changed in Sections 7.7.1.2 (1) (f) and 7.7.1.2 (1) (g) from a "signal" to "signals". This is also consistent with the IED in Figure 7.7-2. This change has no impact on the logic for the initiation of the ARI or SCRRI from the RFCS and changes no design or function of an SSC important to safety. It has no impact on any analyzed accident. This departure has no impact on any SSC system design or function and has no impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

STD DEP 7.7-7, Rod Control and Information System (RCIS) Design Details

Description

Changes were made to the reference ABWR DCD RCIS descriptions in FSAR Subsections 7.7.1.2 (2), (3), (5), and (6), 7.7.1.2.1, and 7.7.1.2.2 to provide clarity, additional information, and provide a more complete design description. The changes addressed the following:

- Description of RCIS monitoring channels.
- “Rod Action Control Cabinet (RACC)” was changed to “Rod Action Control Subsystem (RACS) Cabinets” because the various major subsystem functions were segregated to different cabinets (RAPI panel and ATLM / RWM panel).
- Descriptions of additional RCIS-related panels/cabinets were provided to be consistent with figure 7.7-2 and various major RCIS subsystem functions were allocated into several cabinets.
- Final Remote Communication Cabinet (RCC) implementation details.
- Final Fine Motion Driver Cabinet (FMDC) implementation details.
- Detailed descriptions of the RCIS Multiplexing Network information and the interfaces with class -1E systems are provided
- Detailed descriptions of the RCIS power source are provided.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This change to the STP 3 & 4 FSAR provides a more clear and complete description of the design and operation of the RCIS system. These changes are a result of RCIS design evolution based on experience at operating plants and involve segregation by RCIS subsystems of electronic, electrical, and logic circuitry to different cabinets/panels. This enhanced discussion is not the result of any underlying design change and functional requirements of the RCIS system are unchanged. This departure does not change any Tier 1 or Tier 2* information, the Technical Specifications or Bases or other operational requirements. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no affect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-9, Selected Control Rod Run-In (SCRRI) Function

Description

Subsection 7.7.1.2(6) of the reference ABWR DCD states that the Control Rod Drive (CRD) System provides for electromechanical insertion of selected control rods for core thermal/hydraulic stability control. The STP 3 & 4 FSAR adds that the CRD system also provides for mitigation of a loss of feedwater heating event. This change provided clarity with the additional information and a more complete design description showing the two functional needs for SCRRI.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1, Tier 2* information, the Technical Specifications or their Bases, any underlying design or other operational requirements.

This departure revises Subsection 7.7.1.2(6) to add that the CRD system provides for insertion of selected control rods in response to a loss of feedwater heater event. This function of the SCRRI system is already detailed in the DCD in Sections 7.7.1.2(1)(g) and in 7.7.1.2.2 (2) (b). Consequently, this departure does not change any system design as currently described in the DCD but only updates one particular section, Section 7.7.1.2(6), for completeness and accuracy of the overall documentation. This departure has no impact on any SSC system design or function and has no impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-11, Rod Withdrawal Sequence Restrictions

Description

The STP 3 & 4 FSAR Section 7.7.1.2.1 (5) significantly expands the DCD discussion of the ganged rod movement and ganged withdrawal sequence restrictions. The STP 3 & 4 FSAR provides complete descriptions of these clarifications:

- Provides additional details on the ganged rod mode consisting of one or two sets of fixed control rod gang assignments
- States that the system allows up to 26-rod gangs, for control rods in rod groups 1, 2, 3, and 4, to be withdrawn simultaneously when the reactor is in the startup or run mode

- Revises the maximum allowable difference in rod positions between the leading and trailing operable control rods
- Revises the restrictions on withdrawal of rods in groups

Evaluation Summary

These changes provide an updated design description showing the implemented system design. This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This departure does not change any Tier 1, Tier 2* information, the Technical Specifications or Bases, any underlying design or other operational requirements.

The departure changes as outlined in the description above have been evaluated to determine the impact on the likelihood of accidents previously evaluated. These changes are primarily adding further details to the description of the ganged rod withdrawal restrictions in the STP 3&4 FSAR. This provides for a more complete understanding of the implemented system design. The basic functioning of the RWM of the RCIS to ensure that there exist restrictions on certain ganged control rod movements is unchanged. This departure has no impact on any SSC system design or function and has no impact on the likelihood or consequences of analyzed accidents (e.g. contral rod drop or ATWS) or malfunction of an SSC important to safety. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-12, Rod Control and Information System Indication

Description

This departure updates the discussion of the detailed design of the reference rod pull sequence (RRPS) in STP 3 & 4 FSAR Subsection 7.7.1.2.1 (6). Included in these updates are the following:

- Clarifying that the Plant Computer Function (PCF) and not the Performance Monitoring and Control System (PMCS) is used for storing, modifying and providing compliance verification for the RRPS.
- Clarifying that download of the new RRPS data can only be completed when the RCIS is in manual and when a permissive switch located at the RAPI panel is

activated and not when both keylock permissive switches located at each rod action control cabinet are activated.

- Clarifying that a rod withdrawal block signal (not signals) is generated whenever selected ganged (not single or ganged) rod movements differ from those allowed by the RRPS, when the RCIS is in automatic or semi-automatic rod movement mode.
- Clarifying that the RCIS "activates" an audible alarm instead of "sounding" an audible alarm.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This departure does not change any Tier 1, Tier 2* information, the Technical Specifications or Bases, any underlying design or other operational requirements.

These changes are primarily editorial in nature as noted in the departure description and update and correct several items in the detailed description of the establishment of reference rod pull sequence (RRPS). These changes have no impact on the how the RRPS data is stored, modified or verified. The ability to download the RRPS data using the RCIS is not impacted. Controls continue to be in place to prevent control rod withdrawals when movements differ from those allowed by RRPS. This departure has no impact on any SSC system design or function and has no impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-13, Optical Isolation

Description

This departure removes overly restrictive optical isolation information from the STP 3 & 4 FSAR Subsection 7.7.1.2.1(7) discussion of the Rod Block Function. The change removes the detailed description of the specific type of technology used for optical isolation of rod block signals received by the non-safety Rod Control and Information System (RCIS) from Class 1E systems. The reference ABWR DCD wording discusses the details of a specific technology. The description that all rod block signals from Class

1E systems provided to the RCIS are optically isolated is retained. Also, the description that the optical isolation provides complete isolation while keeping electrical failures from propagating into the RCIS and vice versa is also retained. The retained descriptions adequately cover the requirements for optical isolation of the rod block signals.

Evaluation Summary

This change is deemed necessary to prevent overly restrictive description wording of the type of technology that can be used for achieving suitable optical isolation of the RCIS rod block signals.

This departure has been evaluated pursuant to and determined to comply with the requirements in 10CFR 52, Appendix A, Section VIII B.5. This departure does not change the Technical Specifications, any underlying design or other operational requirements. Furthermore, it does not change the requirements for non-safety system isolation from Class 1E systems, any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses.

This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-14, Rod Control and Information System Bypass

Description

Changes are incorporated in the STP 3 & 4 FSAR Subsection 7.7.1.2.1 discussion of the Rod Control and Information System (RCIS) bypass capabilities:

- Changes to the description of capabilities provided for performing bypass of either the Synchro A or Synchro B position feedback signals for any individual control rod, including the description of RCIS logic that prevents simultaneous bypassing of both synchro signals for an individual control rod.

- Changes in the descriptions regarding the specific location and related operator interface where specific bypass functions can be performed (e.g., update of control rods to be placed in the “Inoperable” status can be performed at the RCIS Dedicated Operator Interface and descriptions of bypass permissive switch for performing certain bypass operations is added for clarity) and the operator interface where bypass status information is available are incorporated.

- Change in the maximum number of control rods that can be placed into the “inoperable” bypass condition only when the reactor mode switch is in REFUEL mode is incorporated (i.e., change required to support control rod maintenance activities during a planned refueling outage nominally every 18 months, instead of nominally every 12 months).

- Changes in the description of the Single/Dual Rod Sequence Restrictions Override (S/DRSRO) bypass to reflect that it is applied to the one or two control rods associated with the same hydraulic control unit (HCU) when performing scram time surveillance testing (and is not a bypass that can be selected for specific individual control rods).
- Addition of a new section to more clearly distinguish the Single Channel RCIS Bypass features from the other RCIS bypass capabilities (i.e., synchro bypass, "Inoperable" bypass, and S/DRSRO bypass are RCIS bypass functions that do not bypass a single channel of the dual redundant RCIS channel equipment). Single Channel RCIS Bypass features are those RCIS bypass functions provided to allow bypass of single channel of dual channel RCIS equipment. The specific list of the available types of Single Channel RCIS Bypass features is also clarified by the changes incorporated.

Evaluation Summary

These changes provide an updated design description showing the implemented system design.This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This change to the STP 3 & 4 FSAR provides a clearer and more complete description of the design and operation of the RCIS system. These changes are a result of RCIS design evolution based on experience at operating plants and involve an enhanced discussion of RCIS bypass capabilies. This departure is not the result of any underlying design change and functional requirements of the RCIS system are unchanged. This departure does not change any Tier 1 or Tier 2* information, the Technical Specifications or Bases or other operational requirements. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no affect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore this departure has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-20, Recirculation Flow Control Logic

Description

Subsection 7.7.1.3 (1) of the reference ABWR DCD describes automatic operation of the Recirculation Flow Control System as only available above 70% power. Subsection 7.7.1.3 (4) provides a more complete description by stating the 70% limit is for a "rod pattern where rated power accompanies 100% flow." This subsection provides further information concerning manual and automatic operation for other rod patterns and power levels. Therefore, the statement “if the power level is above 70% rated” is removed from 7.7.1.3 (1).

FSAR Subsection 7.7.1.3 (4) is further clarified as follows:

- Operation below approximately 25% has been described in lieu of previous information about operation below 70%,
- Load follow capability has been enhanced to include the specific interfacing systems required for this mode of operation in lieu of the original “main turbine regulator control” and
- Terminology for the “main turbine pressure regulator” is changed to “APR” and “semi-automatic mode” is changed to “core flow mode”.

Subsection 7.7.1.3(8)(e) revises the rate limiter rate of change to +5% for increasing speeds and -5% for decreasing speeds, consistent with the speed change rate described in DCD Tier 2 Subsections 15.3.2.1.1 and 15.4.5.1.1.

The terminology is updated in Figure 7.7-5 and Figure 7.7-7 to be consistent with Subsection 7.7.1.3.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This change to the STP 3 & 4 FSAR to clarify and correct inconsistencies regarding operation of the recirculation flow control system does not change any Tier 1 or Tier 2* information, the Technical Specifications or Bases or other operational requirements. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-22, Automated Thermal Limit Monitor (ATLM) Description

Description

The description of the ATLM setpoint and rod block action in reference ABWR DCD Subsections 7.7.1.5 (7) (c) and (7) (e), and 7.7.1.5.1 have been expanded in the STP 3 & 4 FSAR to further describe the interface of interacting systems and this application. The FSAR states that when an ATLM setpoint update is requested, after calculating the power distribution within the core, the computer sends data to the ATLM of the

RCIS on the calculated fuel thermal operating limits and corresponding initial LPRM values. The ATLM monitors various functions and issues rod block signals to prevent violation of the fuel operating limits.

Evaluation Summary

This departure has been evaluated in accordance with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, as described previously. The Automated Thermal Limit Monitor (ATLM) is discussed in Tier 1, the Tech Spec Bases and the LCOs for Control Rod Block Instrumentation. Those sections have been reviewed and the proposed change in the departure has no effect on them. This change is an expansion of the description of the interface of interacting systems and the ATLM. No underlying design change is made and no SSC important to safety or fission product barrier is affected. There is no increase in the frequency of accidents and there is no increase in the likelihood of a malfunction of an SSC important to safety. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-23, Automated Traversing Incore Probe (ATIP) Function

Description

Subsection 7.7.1.5.1 of the reference ABWR discusses inputs from the “automatic fixed incore probe (AFIP)” to be used for gain adjustment factors for Local Power Range Monitoring. The STP 3 & 4 FSAR explains that this function is provided by the ATIP rather than the AFIP in the US ABWR.

Subsection 7.7.1.6.1 (2) of DCD states that the ATIP is nonsafety-related, but the STP 3 & 4 FSAR expands that description to include that this sub-system of the Neutron Monitoring System has no safety function, but the system does contain safety-related components. In response to a containment isolation signal resulting from either low reactor water level or high drywell pressure, the ATIP system automatically initiates TIP probe withdrawal followed by closure of the ball valves and purge line valves to ensure containment isolation.

Subsection 7.7.2.6.2 adds that the ATIP system has isolation valves, and is required to perform automatic containment isolation function in compliance with GDC 56 by following the guidance of Reg. Guide 1.11.

Subsection 7.7.1.6.1 (4) of the DCD states that the ATIP equipment is tested and calibrated using heat balance data and procedures described in the instruction manual. The STP 3 & 4 FSAR states that only the procedures from the instruction manual are required for the calibration of this equipment.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, DCD, technical specifications, basis for technical specifications or operational requirements as a result of these changes.

These items are all clarifying changes which do not involve a change to the ATIP design or function. The ATIP operation is a nonsafety related function which is used to calibrate the LPRM system. Consequently, there is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety.

The ATIP System is not identified as a design feature in the DCD for mitigating an ex-vessel severe accident. These changes to the ATIP system description are clarifying, and therefore the likelihood or consequences of a severe accident are not impacted. As a result of this evaluation, prior NRC approval of these changes is not required.

STD DEP 7.7-24, Steam Bypass and Pressure Control Interfaces

Description

Subsection 7.7.1.8 (7a) of the reference ABWR DCD states that an external signal interface for the Steam Bypass and Pressure Control (SB&PC) System is narrow range dome pressure signals from the SB&PC System to the Recirculation Flow Control System. STP 3 & 4 FSAR Subsection 7.7.1.8 (7a) states that the “narrow range dome pressure signals” are replaced by “validated dome pressure signals.” The signals are validated based on the value of the pressure and the number of signals that are in the valid range.

Based on pressure demand, the SB&PC System calculates position error and flow demand for each turbine valve. Based on these signals, the SB&PC System calculates emergency bypass valve fast opening signal and servo current signal for each turbine valve. Tier 2 Subsection 7.7.1.8 (7h) of the reference ABWR DCD lists these signals as an external interface from the Turbine Bypass System. Tier 2 Subsections 7.7.1.8 (7h) and (7i) of the STP 3 & 4 FSAR list servo current signals as external signal interfaces sent from the SB&PC to the Turbine Bypass System. This clarifies which is the sending unit and which is the receiving unit.

Tier 2 Subsection 7.7.1.8 (7l) of the reference ABWR DCD lists "Governor free demand signal" as an external interface from the APR System. In STP 3 & 4 FSAR Subsection 7.7.1.8 (7l), this signal is replaced by "Automatic Frequency Control signal".

Tier 2 Subsection 7.7.1.8 (7m) of the reference ABWR DCD lists "reactor power compensation signal" as an external interface to the APR System. This list has been changed in Subsection 7.7.1.8 (7m) and (7o) of the STP 3 & 4 FSAR and now it lists "limited speed regulator output" and "pressure regulator output signal" as external interfaces sent to the APR System.

Tier 2 Subsection 7.7.2.8.1 of the reference ABWR DCD states that the SB&PC does not interface with any engineered safeguard or safety related system. STP 3 & 4 FSAR Tier 2 Subsection 7.7.2.8.1 states that the SB&PC System receives reactor pressure and water level from the NBS system but only from nonsafety instrumentation. This clarifies that the SB&PC System does not interface with any safety related instrumentation even though it interfaces with a safety related system.

Figure 7.7-12 and Figure 7.7-13 are also revised to be consistent with the revised description.

Evaluation Summary

The Tier 1 and Tier2* DCD, Technical Specifications, Bases for Technical Specifications and operational requirements were reviewed and were determined not to be impacted by this departure.

This departure clarifies and corrects the description of the I&C interface for the SB&PC System. This departure does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. This departure does not affect any method of evaluation used in establishing the plant design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. This departure does not create a possibility for an accident of a different type than any previously evaluated.

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The change has no adverse impact and does not require prior NRC approval.

STD DEP 7.7-27, RCIS Table Deletion

Description

Table 7.7-1 of the reference ABWR DCD provides the environmental conditions for the Rod Control and Information System (RCIS) module operation environment, consisting of temperature, relative humidity, atmospheric pressure, radiation levels, and seismic acceleration. There is no reference to this table in DCD Section 7.7. or elsewhere in the DCD. DCD Subsection 7.7.1.2.5 references Section 3.11, which provides the requirements for nonsafety-related equipment subject to adverse environments. Therefore, Table 7.7-1 is removed from the STP 3 & 4 FSAR because it is unnecessary.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1 or Tier 2* information, Technical Specifications, any underlying design or other operational requirements. This change is editorial in nature and deletes a table which is not referenced in the DCD and therefore is not necessary. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident or a different type is not created. Also, it does not affect any method used for evaluation in establishing the design or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STP DEP 8.2-1, Electrical Equipment Numbering

Description

Due to the site arrangements of the switchyard and other auxiliary structures, the internal routing of the major power circuits can differ between applicants. Bus assignments and nomenclature may also vary due to site-specific factors. (Also see associated STD DEP 8.3-1 which revised the medium voltage electrical distribution system.) Figure 8.2-1, Sheets 1-7, have been revised to show the new bus numbers and equipment location in the turbine building. Reference subsections were added to the interface requirements to direct the reader to those sections in which the requirements are incorporated.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.This departure to update Figure 8.2-1 to reflect the STP 3 & 4 design for electrical power distribution as described in the FSAR does not change any Tier 1 or Tier 2* information, the Technical Specifications or Bases or other operational requirements. Furthermore, it does not change any plant physical features other than the location of non-Class 1E electrical power distribution equipment. It does not change SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore this change has no adverse impact and does not require prior NRC approval.

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

This departure description and evaluation summary has been moved to Section 2.2.

STD DEP 8A.1-1, Regulatory Guidance for the Lightning Protection System

Description

This departure provides a change from the reference ABWR DCD in STP 3 & 4 FSAR Section 8A.1.2 to acknowledge availability of SRP and regulatory guidance for the lightning protection system. It adds a reference to RG 1.204, November 2005, which is cited in NUREG-0800, Section 8.1, Rev. 3. It also adds references to the applicable sections of IEEE Standards 666, 1050, and C62.23 as they relate to RG 1.204.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This departure does not change any Tier 1, Tier 2* information, the Technical Specifications or Bases, any underlying design or other operational requirements.

This departure updates the lightning protection system requirements from the DCD to reflect the issuance of RG 1.204, which had not been issued at the time of ABWR certification. This RG provides an acceptable approach for the design of lightning protection systems for nuclear power plants. There is no impact on the design or function of an SSC important to safety, and consequently, there is no impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 9.1-1, Update of Fuel Storage and Handling Equipment

Description

Standard Departure 9.1-1 includes the following specific changes:

9.1 Fuel Storage and Handling

The spent fuel storage rack capacity was clarified to be <u>a minimum of</u> 270% to match the description provided in Tier 1 Subsection 2.5.6 and to be consistent with the response to NRC certification question 410.33. In Subsection 9.1.3.3, the maximum 270% has been clarified to be the capacity used for the bounding heat load evaluation. For a pool having a capacity larger than 270%, the additional capacity may not be utilized without revision to the bounding heat load evaluation.

The last paragraph was deleted because the new fuel storage racks are revised for dry storage in a fuel vault and are a different design than the spent fuel storage. This requires different analyses and support structures than for the spent fuel racks. Separate discussions are provided for new fuel storage and spent fuel storage.

9.1.1.1.1 Nuclear Design

Subsection 9.1.6 was changed to Subsection 9.1.6.1 to provide the specific reference number to COL License information.

9.1.1.1.3 Mechanical and Structural Design

The reference to Subsection 9.1.2.1.3 was deleted because it was desired to have a stand-alone description for new fuel storage. Most of the information added to this section previously existed in Subsection 9.1.2.3 and is not a technical change from the DCD. The new or revised information includes the following:

- Changes related to dry vs. wet storage include deleting a liner, and adding a drain and curb to preclude accidental flooding of the new fuel storage racks. Although measures are provided to preclude flooding, COL License Information Item 9.1 (Subsection 9.1.6.1) will demonstrate that K_{eff} is maintained less than or equal to 0.95 with the new fuel racks flooded with unborated water or with the addition of fire extinguishing aerosols.
- The new fuel rack design uses the load combinations described in SRP 3.8.4 Appendix D instead of the DCD description of load combinations.
- The discussion of rack overturning due to horizontal loads was updated and reference to uplift from vertical loads was deleted.
- The supporting gap around new fuel is clarified as being between the fuel and rack instead of between the spent fuel and support tube.

9.1.1.1.4 Thermal-Hydraulic Design

The reference to thermal-hydraulic design for new fuel storage was deleted because it is not applicable to the new fuel dry storage.

9.1.1.1.5 Material Considerations

The reference to Subsection 9.1.2.1.5 was deleted because it was desired to have a stand-alone description for new fuel storage. The information added to Subsection 9.1.1.1.5 previously existed in Subsection 9.1.2.1.5 and is not a technical change from the DCD.

9.1.1.3.2 Structural Design

Anchoring/support details for the new fuel racks were updated to reflect current ABWR practice for new fuel storage.

9.1.1.3.3 Protection Features of the New-Fuel Storage Facilities

Subsection 9.1.1.3.3 is clarified to note that the auxiliary hoist on the Reactor Building crane can be used (vs. is used) for some new fuel movements. The intent is to use the telescoping grapple on the refueling machine for most movements. Refer to Subsection 9.1.4.1 for additional discussion.

A reference to the “rechanneling” area was corrected by substituting “fuel preparation machine” area.

9.1.2.1.2 Storage Design

The spent fuel storage rack capacity was clarified to be a minimum of 270% to match the description provided in Tier 1 Subsection 2.5.6 and to be consistent with the response to NRC certification question 410.33. In Subsection 9.1.3.3, the maximum 270% has been clarified to be the capacity used for the bounding heat load evaluation. For a pool having a capacity larger than 270%, the additional capacity may not be utilized without revision to the bounding heat load evaluation.

9.1.2.1.3 Mechanical and Structural Design

Active vacuum breaker valves in potential siphon paths have been replaced by locating passive vent holes in each pool recirculation line.

The spent fuel rack design uses the load combinations described in SRP 3.8.4, Appendix D, instead of the previous DCD description of load combinations.

Language implying there is one acceptable dynamic analysis method was clarified to permit analysis by different methods if they are approved. Refer also to the COL License Information Item in Subsection 9.1.6.2.

Reference to the AISI code for compressive stability of light gauge structures was eliminated.

9.1.2.1.5 Material Considerations

The missing temperature unit for 16°C was inserted.

9.1.2.3.2 Structural Design and Material Compatibility Requirements

Spent fuel rack anchoring/support details were updated to reflect current ABWR practice.

9.1.3 Fuel Pool Cooling and Cleanup:

9.1.3.1 Design Bases

An acronym for Residual Heat Removal was provided along with the clarification that FPC load is a heat load.

9.1.3.2 System Description:

The word Closed was deleted from “Reactor Building Closed Cooling Water System” for consistency throughout DCD.

The discussion of an RHR loop being available for meeting Mode 4/5 ECCS operability requirements was deleted because this discussion is more appropriately addressed in the Technical Specifications and associated Bases.

The discussion of fuel pool cleanup system performance was supplemented and clarified to add suspended solids removal capability, total corrosion product metal below 30 ppb, and a flow rate of two water changes per day.

9.1.3.3 Safety Evaluation

In previous sections, the spent fuel storage rack capacity was clarified to be a minimum of 270% to match the description provided in Tier 1 Subsection 2.5.6 and to be consistent with the response to NRC certification question 410.33. In this section the maximum 270% has been clarified to be the capacity used for the bounding heat load evaluation. For a pool having a capacity larger than 270%, the additional capacity may not be utilized without revision to the bounding heat load evaluation.

Clarified that the makeup water supply to the fuel pool is from the Makeup Water Condensate System (MUWC).

Clarified the description of the valve arrangement for isolating the non-seismic filter-demineralizers to be consistent with Figure 2.6.2 of the Tier 1 DCD.

Updated the COL License Item in Subsection 9.1.6.9.

9.1.4 Light Load Handling System (Related to Refueling)

Changes in this subsection were made to update the equipment and special tools utilized in ABWR refueling operations, including the inspection of new fuel. The Refueling machine is described as Seismic Category I. Outdated equipment (e.g. vacuum sipper) that is no longer utilized was deleted. The specific changes are noted in Subsection 9.1.4 and include the following:

(a) Revised the channel bolt wrench size.

(b) Revised the channel bolt wrench size.

(c) Deletion of the Fuel Vacuum Sipper and restoration of the Fuel Assembly Sampler.

(d) Renamed some of the reactor vessel service tools.

(e) Clarified the description and operation of the RPV head strongback and stud tensioner.

(f) Revised the capacity of one refueling machine auxiliary hoists.

(g) Deletion of the use of In-vessel Rack.

(h) Identified the maximum speed of the refueling machine grapple hoist.

(i) Upgraded the steamline plug description to include materials and seals.

(j) Upgraded the shroud head stud wrench description of materials.

(k) Removed the description of the blade guide weight.

(l) Corrected the weight and dimensions of the RIP motor.

(m) Revised the description of the general purpose grapple.

(n) Removed the spring reel for incore servicing.

(o) Removed the incore flange seal test plug.

Additionally, Subsection 9.1.4 is updated to clarify the use and operation of the various equipment and special tools.

Other changes in this subsection not specifically related to equipment or special tools involved in refueling are:

9.1.4.1 Design Bases

Clarified that the minimum water level for shielding is the height above the top of active fuel (TAF).

9.1.4.2.1 Spent Fuel Cask

Revised this subsection to reflect that information related to a spent fuel cask will be the subject of future updates to the FSAR when movement of fuel from the spent fuel pool becomes necessary.

9.1.5 Overhead Heavy Load Handling Systems (OHLH)

Changes were made to update OHLH utilized in ABWR refueling operations, including the addition of ASME NOG-1 as a technical standard for the Type I Reactor Building crane. The description and use of the under vessel rotating platform was also updated. The specific changes are noted in Subsection 9.1.5 and summarized as follows:

(a) Updated the RIP motor weight and dimensions.

(b) Removed the in-vessel rack.

14.2.12.1.50 Fuel-Handling and Reactor Component Servicing Equipment Preoperational Test

The flange seal test plug is replaced with sealing equipment to be consistent with the information in Subsection 9.1.4.

Tables and Figures

Tables 9.1-1, 9.1-9, 9.1-10 and 9.1-12 were not changed. The remaining tables were revised to reflect revisions made in the text and other additional changes as shown. Table changes are presented as new tables with all changes incorporated.

Table 1.8-21~~a~~ was revised to reflect the 2004 edition of ASME NOG-1.

Figures 9.1-4, 9.1-7, and 9.1-10 are deleted.

Figures 9.1-1, 9.1-2, 9.1-5, 9.1-8, 9.1-11, 9.1-12 and 9.1-14 are revised.

Evaluation Summary

This departure has been evaluated pursuant to and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The changes to fuel handling equipment and special tools are essentially material, weight and dimensional changes, corrections or clarifications (e.g., for wrenches and grapples, Steamline Plug, RIP motors, Blade Guide), use of the Fuel Assembly Sampler instead of the Fuel Vacuum Sipper, design changes to the RPV Stud Tensioner System, and Refueling Machine Auxiliary Hoist (for increased capacity) to adapt previous ABWR or other BWR plant experience, the addition of sealing to the Steamline Plug for more effective and reliable sealing, and the deletion of the In-Vessel Rack to minimize potential for objects to drop in the reactor during temporary moves.

This departure does not change the Technical Specifications, or other operational requirements of the fuel-handling operation. Furthermore, it does not change any plant physical features other than fuel-handling equipment and tools, and does not affect any SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 9.2-1, Reactor Building Cooling Water System

Description

This departure revises the Design Characteristics Table 9.2-4d as follows: The Reactor Building Cooling Water (RCW) System heat exchanger design capacity for divisions A and B is increased from 47.73 GJ/h to 50.1 GJ/h each, and division C is increased from 44.38 GJ/h to 46.1 GJ/h. The increased RCW heat exchanger design capacity values are based on meeting the LOCA heat loads with a performance margin of 20% to allow for fouling. This also provides a greater heat removal capability during RHR operation. These changes represent a conservative increase in the ability of the system to perform its safety and power generation heat removal design functions.

This departure also corrects inconsistencies in the System Description in Section 9.2.11.2. The design heat removal capability of each division of the RCW from RHR was corrected in the text from 107.6 GJ/h to 108.02 GJ/h to make it consistent with the numbers already provided in Tables 9.2-4a, 9.2-4b and 9.2-4c. As a result of that change, the amount of sensible heat removed by each division with 3 divisions operating was revised in the text from 63.2 GJ/h to 63.62 GJ/h. Also as a result of that change, the amount of sensible heat removed by each division with 2 divisions operating was revised from 41.0 GJ/h to 41.42 GJ/h.

In addition, a clarification is made in Section 9.2.11.3.2 that all heat exchangers and pumps “are normally placed in operation” as opposed to “will be required” for shutdown cooling. Previous discussions in that section showed that shutdown cooling can be performed with less than all of the pumps and heat exchangers operating.

Evaluation Summary

This departure has been evaluated in accordance with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This change increases the heat exchanger design capacity for the Reactor Building Cooling Water (RCW) System for all 3 divisions of this system. As a result, this change will provide a greater heat removal capability during RHR operation both for normal shutdown cooling and in response to a LOCA event. Greater margin is being provided between the decay and sensible heat generated in the reactor core following a LOCA event and the capability of the cooling system to remove the heat. The change in capacity does not introduce new equipment nor does it affect system redundancy. Essential plant cooling can still be met by the remaining operable heat exchangers should one fail and as such this change will not result in an increase in the frequency or consequences of an accident or malfunction of an SSC important to safety. No design basis limit for a fission product barrier is being exceeded or altered by this departure. There is no impact on any Tier 1, Tier 2* DCD, technical specifications, bases for technical specifications or operational requirements as a result of this departure. Consequently, prior NRC approval is not required.

STP DEP 9.2-2, Makeup Water Preparation System

Description

Changes specific to the operation of the Makeup Water Preparation (MWP) System are:

- The flow capacity of each division of the MWP System has been doubled from 45 m^3/h to 90 m^3/h.
- The storage capacity for demineralized water has been increased from 760 m^3 to at least 5320 m^3.
- The MWP System is capable of providing demineralized water at the reference ABWR DCD specified flow rate of 135 m^3/h per unit, but for periods of short duration. Average sustainable flows will be lower as needed to meet demands.
- The MWP System is designed to supply makeup water to the Ultimate Heat Sink (UHS) basin and the Fire Protection system on an as needed basis. New interfaces to the UHS basin and the Fire Protection System provide an additional makeup water supply. The MWP system interfaces do not serve any safety functions. The UHS and Fire Protection System are designed with adequate storage to serve all their safety related functions without the supply of makeup water.
- The capacity of the MWP System to provide water to the Potable and Sanitary Water System has been doubled from 45 m^3/h to 90 m^3/h for instantaneous flows and the source is well water, not filtered water. The potable water is supplied unfiltered directly from the wells in accordance with state and local codes and regulations. This is consistent with the STP 1 & 2 Potable Water System.
- Demineralized water prover tanks have been added to increase the storage capacity and monitor water quality and sulfuric acid chemical feed tanks have been added to further reduce fouling and scaling in the reverse osmosis filter membranes.

This is another dual unit change such that the MWP System is capable of supplying both STP Units 3 & 4. See STP DEP 1.1-2 for a discussion of shared structures, systems and components between STP 3 & 4.

Increased capacities of various tanks/basin provide more flexibility to accommodate peak demands for the MWP system.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change.

The MWP System is described in the reference ABWR DCD and is not designated as equipment important to safety. The basic function of the MWP System as described by the DCD is not significantly changed by this departure nor do these changes affect interactions with previously designated equipment important to safety. Changing the description of the source of water to the Potable and Sanitary Water System, well water that is not filtered, and clarifications to the characteristics of MWP System equipment does not affect critical parameters associated with the cause of a previously described accident or malfunction of equipment important to safety. The MWP System is not relied upon to mitigate the consequences of accidents or equipment important to safety. Changes associated with this departure do not affect fission product barriers. These changes do not affect the probability of occurrence of a severe accident as described by the DCD, nor do they increase the consequences of a severe accident.

Based on the evaluation, prior NRC approval of the change is not required.

STP DEP 9.2-3, Turbine Building Cooling Water System

Description

The heat removal capacity of each of the three heat exchangers in the Turbine Building Cooling Water System is increased from 68.7 GJ/h to 114.5 GJ/h and the flow rate of each of the three pumps is increased from 3405 m^3/h to 4550 m^3/h due to increased heat loads caused by alterations to Turbine Island equipment (e.g., number of pumps and increased non-essential chiller size).

The following heat loads are changed from DCD:

- Additional Heat Loads
- Generator H_2 gas dryer cooler
- Offgas condensers
- Condensate booster pumps
- Sample coolers

Deleted Heat Loads:

- Generator breaker coolers
- Seal oil coolers
- Exciter coolers

Modified Equipment Quantities:

- Mechanical vacuum pump seal water cooler: from one to two
- Heater drain pump oil and motor air cooler: from two to four

- Reactor feedwater pump oil, motor air, and ASD coolers: from three to four
- Iso Phase Bus coolers: from two to three

Relocated

- Lube oil temperature control valve from cooler downstream to upstream

Evaluation Summary

The change discribed above is limited to the heat load capacity of the TCW equipment. This change does not affect the safety analyses as discussed below:

- The systems are all nonsafety-related. Therefore, they provide no safety functions and are not used to mitigate the consequences of any accidents.
- The departure does affect any safety system, structure and component. The departure does not cause an increase in dose exposure to public.
- The departure is not used as the assumption for plant transient analysis nor safety analysis. Therefore, the departure does not cause an increase in the dose exposure to public.
- The departure does not increase the consequence of a malfunction of systems, structures, and components important to safety.
- The departure will not cause any accident of a different type than evaluated previously in the reference ABWR DCD. There is no impact on the probability or consequences of an accident of malfunction of an SSC important to safety. This departure does not cause an increase in the dose exposure to public.

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, as described previously. The change has no adverse impact, and affects the function, but is bounded by the safety analysis.

There is no impact on any Tier 1 or Tier 2* DCD, technical specifications, or operational requirements as a result of these changes. Consequently, prior NRC approval is not required for this departure.

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

Description

During preparation of the STP 3 & 4 FSAR, it was identified that the RSW flow rate specified in the reference ABWR DCD needs to be increased in order for the site-specific RSW system to accomplish its safety and power generation design bases. Increasing the RSW flow rate results in the following changes to Subsection 9.2.15:

- RSW system pipe sizes in Figure 9.2-7, Sheets 1-3, are increased
- RSW flow rate per pump in Table 9.2-13 is increased from 1,800 m^3/h to 3,290 m^3/h

RSW heat exchanger flow (heat removal requirement) has been increased due to the heat load from RCW System. This flow will be accommodated by higher capacity pumps and larger pipe diameter.

This departure also adds in-service testing requirements for the system valves as shown in Table 3.9-8.

Additionally, this departure is affected by the change in configuration of the UHS. The change of the UHS results in the following changes to Subsection 9.2.15:

- In Table 9.2-13, "Pump total head 0.34 MPa" is changed to "Pump total discharge pressure 0.67 MPa".
- In Table 9.2-13, "Design pressure 0.79 MPa" for RSW pump is changed to "Max. operating pressure 1.42 MPa".
- Design pressure of RSW piping and valves in Table 9.2-13 is increased from 1.08 MPa to 1.56 MPa.

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

Evaluation Summary

This departure does not change any Tier 1, Tier 2* information, the Technical Specifications or Bases, or other operational requirements.

This departure updates the STP 3 & 4 FSAR to provide site specific information on the RSW system in-service testing requirements. It further increases the heat removal capability of this system relative to that specified in the DCD. The individual pump flow rate was increased from 1800 m^3/h to 3290 m^3/h, "Pump total head, 0.34 MPa" was changed to "Pump total discharge pressure, 0.67 MPa", "Design pressure, 0.79 MPa" for RSW pump was changed to "Max. operating pressure, 1.42 MPa" and design pressure of RSW piping and valves was changed from 1.08 MPa to 1.56 MPa in Table 9.2-13. Piping sizes were also increased to accommodate the higher flow rate. This additional heat removal capability is required because of the additional RCW heat exchanger design heat removal capacity as identified in Departure STD DEP 9.2-1. That additional RCW capacity was based on meeting the LOCA heat loads with a performance margin of 20% to allow for fouling. As noted in the evaluation of that departure, the increased capacity of the RCW provides a greater heat removal capability during RHR operation both for normal shutdown cooling and in response to a LOCA event. Greater margin is being provided between the decay and sensible heat generated in the reactor core following a LOCA event and the capability of the cooling system to remove that heat and as such this is a favorable change.

On the other hand, increase in the RSW pump flow rate and diameter of the discharge piping has a potential to adversely impact the Control Building flooding analysis. Separation of each division of the RSW heat exchanger room and interlocks to trip the RSW pump and to close the motor-operated valves F013 and F014 by the flooding level high signal mitigate the impact. Detail is indicated in COL License item 19.9.26. Consequently, this departure has no adverse impact on any SSC system design or function and has no adverse impact on the likelihood or consequences of analyzed accidents except the flooding or malfunction of an SSC important to safety. Furthermore, it does not change any fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 9.2-7, HVAC Normal Cooling Water System

Description

This departure reflects a design change to correct inconsistencies in reference ABWR DCD Tables 6.2-9, 9.2-6, 9.2-7, 9.4-1, and Figure 9.2-2 such that the nonsafety-related HVAC Normal Cooling Water (HNCW) system waterside heat removal rate is greater than or equal to the airside cooling duty heat loads. The capacity and flow rate for each HNCW chiller are also increased to include the revised heat loads. This will ensure that the waterside heat removal design capacity is sufficient to remove the heat load from the airside. Note that departure 9.2-9 increased the chilled water return temperature to help offset the impact of departure 9.2-7 on increased HNCW equipment size. Therefore the changes made in the Tables affected by departure 9.2-7 also included the effect of departure 9.2-9 with the exception of chiller cooling capacity and condenser water flow per unit, in Table 9.2-6.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This design change from the ABWR DCD reflects changes to the HVAC Normal Cooling Water (HNCW) System to ensure there is sufficient heat removal capability for the revised heat loads. This change to the STP 3 & 4 FSAR to reflect the HNCW design does not change any Tier 1 or Tier 2* information, the Technical Specifications or Bases or other operational requirements. Furthermore, it does not change any plant physical features other than the HNCW system. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This change modifies the HNCW line size, which penetrates the secondary containment, which is a fission product barrier. However, the small increase (by 50 mm) of the penetration size does not alter the design basis limit for this fission product barrier. The small diameter increase has a minimal effect on the secondary containment stresses, and does not result in the design basis limit being exceeded.

This departure does not adversely affect the function of any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no affect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STP DEP 9.2-8, Potable and Sanitary Water System

Description

This departure makes the following changes specific to the operation of the Potable and Sanitary Water (PSW) System:

- The minimum flow capacity of the PSW System has been doubled from 45 m^3/h to 90 m^3/h to ensure that potable water is provided to all buildings at STP 3 & 4, intermittently, during peak demands.
- Potable water is not filtered, is supplied directly off of the well water system, and is chemically treated to prevent harmful physiological effects on plant personnel.
- The Sewage Treatment System, including the Sanitary Drainage System, is provided and sized to treat sanitary waste for all four units at the STP site.

This is another dual-unit change such that the potable water subsystem is capable of supplying both STP 3 & 4 and the sewage treatment subsystem is capable of treating sanitary wastes collected from all four units located at the site. This increase in system capacity ensures flexibility and reliability for future needs at the site.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change.

The PSW System is described for the standard ABWR plant in the DCD and is not addressed as equipment important to safety, The basic function of the PSW System as described by the DCD is not significantly changed by this departure nor do these changes affect interactions with previously designated equipment important to safety. Changing the description of the source of water to the Potable and Sanitary Water System well water that is not filtered, and clarifications to the characteristics of PSW System equipment does not affect critical parameters associated with a previously described accident or malfunction of equipment important to safety. The PSW System is not relied upon to mitigate the consequences of accidents or equipment important to safety. Changes associated with this departure do not affect fission product barriers. These changes do not affect the probability of occurrence of a severe accident as described by the DCD, nor do they increase the consequences of a severe accident.

Based on the evaluation, prior NRC approval of the change is not required.

STP DEP 9.2-9 HNCW Cooling Water System

Description

This departure changes the HNCW return temperature from 12°C to 14.7°C and modifies the HNCW equipment to reduce equipment, piping, valve sizing and electrical power for better maintainability. This departure affects Subsection 9.2.12 (Tables 9.2-6 and 9.2-7), Table 9.4-1 and Table 6.2-9. Note that in Table 9.2-6, the increased chilled water delta T only impacted the chilled water flow rate and pump capacity of 560 cubic meters/hour.

This change is closely related to departure 9.2-7 which tended to increase the chilled water flow rates to equipment cooled by HNCW. By raising the chilled water return temperature, departure 9.2-9 offset the increase in chilled water flow rate that would have resulted from departure 9.2-7 by itself and permitted the resulting adverse impact on equipment size to be minimized.

Evaluation Summary

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2* and technical specifications.

The departure increases the HNCW return temperature. As indicated above, by raising the chilled water return temperature, departure 9.2-9 offsets the increase in chilled water flow rate that would have resulted from departure 9.2-7 by itself and permitted the resulting adverse impact on equipment size to be minimized.Therefore, this change has minimal effect on the frequency or consequences of accidents, or the probability of consequences of malfunctions. The systems involved are not relied upon for ex-vessel severe accident mitigation.

As a result, prior NRC approval of this departure is not required.

STP DEP 9.2-10, Turbine Service Water System

Description

The Turbine Service Water (TSW) pumps transfer Main Cooling Reservoir water for the Turbine closed cooling water heat exchangers installed inside of the turbine building. The TSW System design parameters are revised to reflect site specific information. These changes include the TSW pump head and discharge flow, the TSW system design pressure, the location of the TSW pump house, the temperature increase and pressure drop across the Turbine Cooling Water (TCW) heat exchangers, and the number of TCW discharge lines. A filling line is also added to the TSW pump discharge, and the TSW system inlet and outlet are modified to reflect that these lines come from and go to the main cooling reservoir. This departure impacts FSAR Subsection 9.2.16.2, Tables 9.2-16, 19R-1, and Figure 9.2-8.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of these changes. The TSW system is a nonsafety-related system. The proposed changes to the Turbine Service Water system proposed in this departure are to meet site specific conditions. The proposed changes do not impact the design or function of any SSC important to safety as a result of this change. Furthermore, the TSW system is not used to mitigate any accident. As a result of this departure, there is no effect on the frequency or consequences of any accidents or the likelihood or consequences of malfunctions of SSC important to safety previously evaluated in the DCD. There is no possibility of a new type of accident, and there is no impact on fission product barriers or ex-vessel severe accident events.

Therefore, the change has no adverse impact and does not require prior NRC approval.

STD DEP 9.3-1, Radwaste Drain Materials

Description

This departure replaces the carbon steel piping in the Radwaste Collection System with stainless steel piping. Due to the greater corrosion resistance of stainless steel compared with carbon steel, this change will significantly reduce the amount of contaminated corrosion products, the load on the liquid radwaste system, and the solid radwaste shipment volume. This is consistent with NRC and industry initiatives for radwaste volume reduction.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

The Tier 1 and Tier 2* DCD, technical specifications, basis for technical specifications and operational requirements were reviewed and were not impacted by this change.

As noted above, this departure is a design improvement in the materials specified for the radwaste piping which will reduce the volume of liquid and solid radwaste. This is a favorable change which will not adversely affect the likelihood or consequences of analyzed accidents or malfunction of SSC important to safety. There is no change to any design basis for a fission product barrier nor are there any new accident scenarios created.

The radwaste collection system has not been identified as a design feature in the DCD for mitigating an ex-vessel severe accident. In addition this is a design improvement which will reduce the volume of liquid and solid radwaste. Therefore, there is no increase in the probability or consequences of an ex-vessel severe accident.

Based on the results of this evaluation, prior NRC approval is not required.

STD DEP 9.3-2, Breathing Air System

Description

For industrial health reasons and as recommended by the Utility Requirements Document, this departure provides a new breathing air system (BAS) that is entirely separated from the existing service air system (SAS) as described in the reference ABWR DCD. The system consists of a bottled breathing air supply and a portable or permanent breathing air compressor supply system as-needed. The BAS will supply the Turbine Island, Nuclear Island and Radwaste Building. Backup to the BAS will consist of dedicated breathing air bottles.

Also, during preparation of the departure, inconsistencies in the DCD for the SAS were noted and corrected. Specifically, the SAS supply to containment is provided with a containment isolation scheme meeting the requirements of general design criterion (GDC) 56 “Primary containment isolation.” The isolation scheme is a check valve inside and a locked closed manual globe valve outside containment (GDC 56, Option (2)). However in a number of places the DCD inconsistently refers to the penetration as being a GDC 57 “Closed system isolation valves” penetration, or refers to the SAS as a closed system, or refers to the inside containment isolation check valve as a globe valve. These inconsistencies are corrected.

Like the SAS, breathing air is supplied inside containment only while shutdown. This departure will utilize an existing spare containment penetration to supply the BAS inside containment and equips the penetration with two locked closed manual globe valves meeting the requirements of GDC 56, Option (1). Like SAS and the instrument air system (IAS), the new BAS penetrates secondary containment but also like SAS and IAS, the primary containment isolation scheme for BAS precludes it from being a potential bypass leakage path through the secondary containment.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1 or Tier 2* DCD, Technical Specifications, basis for Technical Specifications or operational requirements as a result of this change.

For the SAS, this change makes editorial corrections (corrects inconsistencies) with no change to meaning or intent. The SAS was clearly intended to be considered a GDC 56 type penetration with two series containment isolation valves rather than a GDC 57 penetration with a single isolation valve. These inconsistencies have been corrected.

For the BAS this change consists of technical changes to the ABWR DCD pertaining to the separation of the BAS from the SAS.

The air supplied by the SAS and the BAS does not have a safety related use; however, the containment penetrations associated with the SAS and BAS have a safety related function to maintain containment integrity under design basis accidents. The safety significance of this departure is the change of a spare penetration to an additional functional containment penetration. However the same technical requirements that were imposed by the DCD on the BAS (as an integral part of the SAS) are now imposed by the FSAR on the new separate BAS. In addition, the locked closed containment isolation valves associated with the new BAS penetration are subject to the same locked valve administrative controls previously applied to the SAS inboard globe valve. Therefore the BAS containment penetration does not contribute to more than a minimal increase in the frequency of occurrence of an accident, nor an increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the ABWR DCD. The changes to the BAS do not affect the probability of the occurrence of a severe accident as described by the DCD, nor do they increase the consequences of a severe accident.

Based on this evaluation, prior NRC approval of the change is not required.

STD DEP 9.3-3, Control Rod Drive System Sampling

Description

ABWR DCD Subsection 9.3.2.3.1 and Table 9.3.2 describe measurements of Oxygen and Conductivity for Control Rod Drive(CRD) System water. These water qualities can be monitored by Condensate Purification System (CPS) effluent and removal of these measurements has no safety significance. Therefore, these are integrated with measurement of Oxygen and Conductivity for CPS water. To keep the consistency of measurements between CPS and Feedwater, instrument range of the CPS effluent is changed from “0 to 100 ppb” to “0 to 250 ppb” and high alarm setpoint “200 ppb” is added to the CPS effluent.

Evaluation Summary

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, as described previously.

It does not change the functional or safety requirements of the CRD system. The CRD water sources are the effluent from the CPS and the Condensate Storage Tank (CST). During plant operation, CRD water source is the effluent from the CPS. Water qualities of CRD system and CPS effluent are same. The changed instrument range is consistent with that of Feedwater. And the added high alarm setpoint is consistent with those of the CRD system and Feedwater. Furthermore, this change does not impact any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STP DEP 9.4-1, Service Building HVAC System

Description

This site-specific departure modifies the reference ABWR DCD for the Service Building HVAC System. It revises the outside inlet air monitoring instrumentation design by removing the provisions for toxic gas monitors and the Technical Support Center (TSC) alarm for high toxic gas concentration. The toxic gas monitors and the TSC alarm can be deleted from the design based on the site-specific evaluation of on-site and off-site mobile and stationary sources of toxic gases described in FSAR Subsection 2.2S in accordance with Regulatory Guide 1.78.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change.

The Service Building HVAC System toxic gas monitors and alarms are not safety-related but are important to safety. The toxic gas monitors and alarms from the system are not needed based on the analysis contained in the FSAR Section 2.2S and their elimination does not affect the operation of the Service Building HVAC System. The basic function of the Service Building HVAC System as described by the DCD is not significantly changed by this departure and this change does not affect interactions with previously designated equipment important to safety. Elimination of the toxic gas monitors and alarms from the Service Building HVAC System does not affect previously described accidents or malfunction of equipment important to safety. The toxic gas monitors and alarms are not needed to mitigate the consequences of accidents or malfunctions of equipment important to safety. Changes associated with this departure do not affect fission product barriers. These changes do not affect the probability of occurrence of a severe accident as described by the DCD, nor do they increase the consequences of a severe accident.

Based on this evaluation, prior NRC approval of the change is not required.

STD DEP 9.4-2, Control Building HVAC System

Description

This standard departure provides for changes from the reference ABWR DCD of the smoke removal mode of operation of the Control Building HVAC System described in FSAR Subsections 6.4.4.2, 9.4.1.1.4 and 9.5.1.1.6 as described below:

- FSAR Figure 9.4-1, sheets 1 and 2 are revised to include a control room main air supply duct bypass line around the air-handling unit with two motor operated dampers for each of the two control room habitability area HVAC divisions.

- FSAR Figure 9.4-1, sheets 3 through 5 are revised to include a control building air supply bypass line with two motor operated dampers around the air-handling unit in each of the three safety-related equipment HVAC areas.
- FSAR Sections 6.4.4.2, 9.4.1.1.4 and 9.5.1.1.6 are revised to describe how the dampers (described in the two bullets above) operate during the smoke removal mode.

Each air supply bypass line and damper arrangement as described above is required to provide a balanced air flow such that smoke is exhausted and not transported into other areas of the control building. This air balance during smoke removal mode of operation is required because of the large mismatch between the air inlet supply (80,000m^3/h) and the air exhaust (10,000 m^3/h total; 5,000 m^3/h for each exhaust fan).

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, technical specifications, bases for technical specifications or operational requirements as a result of these changes.

Providing the Control Building HVAC Air Handling Units with bypass lines and associated dampers provides the necessary assurance that smoke is exhausted and not transported to other areas of the Control Building. As such, this change provides equivalent or better smoke removal, and thus does not have any effect on the frequency of occurrence or consequences of accidents or malfunction of SSC important to safety previously analyzed.

This change involves the design of the control room HVAC to assure air balance during smoke removal; this has no effect on the probability or consequences of an ex-vessel severe accident.

As a result of this evaluation, prior NRC approval of the change is not required.

STD DEP 9.4-3, Service Building HVAC System

Description

The Service Building HVAC System described in the reference ABWR DCD had two subsystems, the Clean Area HVAC System and the Controlled Area HVAC System. This standard departure described in STP FSAR Subsection 9.4.8 deletes the subsystems and consolidates the Service Building HVAC System to supply air to both the Clean Area and the Controlled Area. The Service Building HVAC System is included as a load powered by the Combustion Turbine Generator that can be manually loaded by the operator. This allows the Technical Support Center and Operations Support Center to be habitable under accident conditions.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change.

The proposed changes to the Service Building HVAC System include descriptive changes from two HVAC subsystems to one HVAC system that are consistent with the functional description of the system currently described by the ABWR DCD. An additional provision is included with the proposed change under this departure to make it possible for operation of Service Building HVAC using power from the Combustion Turbine Generator during loss of offsite power conditions. The basic function of the Service Building HVAC System and Combustion Turbine Generator as described by the DCD is not significantly changed by this departure and this change does not affect interactions with previously designated equipment important to safety. These changes do not affect a previously described accident or malfunction of equipment important to safety. The proposed changes are an enhancement to the ability to operate the Service Building HVAC System using power supplied by the Combustion Turbine Generator during a loss of offsite power event. Changes associated with this departure do not affect fission product barriers. These changes do not affect the probability of occurrence of a severe accident as described by the DCD, nor do they increase the consequences of a severe accident.

Based on this evaluation, prior NRC approval of the change is not required.

STD DEP 9.4-4, Turbine Island HVAC System

Description

This departure revises STP 3 & 4 Turbine Island HVAC system flow rate and cooling/heating load to accommodate the effect of change of the Turbine Building General Arrangement and systems located in Turbine Building including the following changes specified in other departures:

- STP 3 & 4 turbine generator has been changed. (STD DEP 10.2-1)
- The power generation heat sink described in the DCD (natural draft cooling tower) is being replaced by a cooling reservoir. (STD DEP 10.4-2)
- The DCD medium voltage electrical system design is being replaced by a dual voltage design.
- STP 3 & 4 Off Gas system charcoal adsorber vault in OG Holdup room temperature range adopts newest design.(STD DEP 11.3-1)
- Various other departures which change the quantity and arrangement of equipment in the Turbine Building. (e.g. STD DEP 10.4-5 for increased number of Feed Water Pumps.)

This departure adds the OG Holdup room temperature range limits to Subsection 9.4.4.1.2. Also, subsections 9.4.4.2.1.5 and 9.4.4.2.2.2 are revised to explain that local unit coolers and heaters are installed in high heat load areas, and the descriptions of specific areas with local unit coolers are deleted since the list of those areas is added to Figures 9.4-2b (Sheet 2) & c. The system and equipment specifications in Tables 9.4-3 and 9.4-5, and Figure 9.4-2a are changed as a result of a heat load re-calculation for the revised Turbine Building General Arrangement and Off Gas System requirements. The departure changes the HNCW water load described in Table 9.2-7 due to the changes in the Turbine Island HVAC.

STP 3 & 4 nonsafety-related electrical equipment is installed in a non-radioactive controlled area of Turbine Building, and ventilating and air-conditioning of these areas is performed by the Turbine Building Electrical Equipment Area (TBEEA) HVAC. Thus, it is necessary to change the Turbine Island HVAC subsystem name in section 9.4 from Electrical Building (E/B) HVAC to Turbine Building Electrical Equipment Area (TBEEA) HVAC.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of these changes.

This departure changes STP 3 & 4 Turbine Island HVAC system flow rate and cooling/heating load based on revised Turbine Building General Arrangement and Off Gas System requirements. Since this change does not affect any other plant SSCs, there is no effect on any accident previously evaluated in the DCD. Furthermore, although changes are made to certain plant physical features as described above, SSCs important to safety and fission product barriers are not affected in any way. Therefore, previously evaluated accidents are not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, the change has no adverse impact and does not require prior NRC approval.

STD DEP 9.4-5, Radwaste Building Ventilation

Description

This standard departure aligned the system described in the FSAR text with the figures depicting the system and eliminated HVAC equipment supporting the radwaste incinerator, which was deleted. The radwaste control room HVAC description was modified to be consistent with Figure 9.4-10 and the description of control room systems operation was clarified to demonstrate proper control room boundary pressurization.

A dedicated air conditioning system for electrical, HVAC equipment rooms and other areas was added as a result of design evolution.

Operation control of the exhaust air system from the radwaste process areas is augmented to automatically route the exhaust air through the filtration equipment upon detection of airborne radioactivity in the exhaust airflow, this will provide control of radioactivity release from the building and also reduces the replacement frequency of the filter banks of the air filtration equipment.

Evaluation Summary

The changes to the Figure 9.4-10, sheets 1,2, & 3 were based on calculations and site specific general arrangements and temperatures.

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, as described previously. The change has no adverse impact, and affects the function, but is bounded by the safety analysis.

Although this departure alters the physical plant by adding a dedicated air conditioning system for electrical, HVAC equipment rooms and other areas, and thus is a change to the plant physical features, SSCs important to safety and fission product barriers are not affected. Therefore, previously evaluated accidents are not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any necessary feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 9.4-6, Control Building HVAC System

Description

The reference ABWR DCD, Tier 2 Subsection 9.4.1, contains one flow element/flow switch (FE/FS) in the common discharge duct of each emergency filtration unit which is used to automatically start the standby emergency filtration unit in the event of operating unit low flow or fan failure. This departure changes the number, location and logic of these FEs/FSs. Instead of one FE/FS per division installed in the common discharge duct, a FE/FS is to be installed on the discharge side of each emergency filtration unit fan (two fans per redundant division, 4 total for the Control Room Habitability Area(CRHA) HVAC System, as depicted in Figure 9.4-1, Sheets 1 and 2. Within each redundant division, a two-out-of-two logic signal is required to automatically initiate switchover to the standby division. Utilization of 2 FEs/FSs in this manner places the system in conformance with Technical Specification 3.3.7.1.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, technical specifications, bases for technical specifications or operational requirements as a result of these changes.

Providing two channels per emergency filtration (EF) division instead of one, with two out of two logic for switchover to the standby division, provides added assurance that switching to the standby division only occurs when the primary EF has low flow in both fan units. This change provides equivalent or better EF operation in response to emergency conditions, thus does not have any effect on the frequency of occurence or consequences of accidents or malfunction of SSC important to safety previously analyzed.

This change involves the operation and switching of the primary and redundant emergency filtration divisions for the CRHA; these have no effect on the probability or consequences of an ex-vessel severe accident.

As a result of the evaluation, prior NRC approval of the change is not required.

STD DEP 9.4-7, Control Building HVAC System

Description

The reference ABWR DCD states that the MG sets are located in the CB and the MG Set rooms are ventilated by CB Safety-Related Equipment Area (CBSREA) HVAC and cooled by MG Set Room AHUs. This standard departure addresses the requirements for ventilation of the Control Building Annex (CB Annex) due to moving the Reactor Internal Pump (RIP) motor generator (MG) sets from the Control Building (CB) to the CB Annex. This departure modifies the CB Annex HVAC to provide appropriate ventilation, filtering cooling and heating of the MG Set rooms in the Control Building Annex.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, bases for technical specifications or operational requirements as a result of these changes.

Revising the design of the Control Building Annex HVAC system to appropriately accommodate the MG sets relocated from the control building provides similar HVAC performance for this added nonsafety-related equipment, thus does not have any effect on the frequency of occurrence or consequences of accidents or malfunction of SSC important to safety previously analyzed.

This change involves systems which are not relied upon for mitigation of ex-vessel severe accident. Therefore the likelihood or consequences of an ex-vessel severe accident is not impacted.

As a result of this evaluation, prior NRC approval of the change is not required.

STP DEP 9.4-8, Reactor Building HVAC

Description

Tornado dampers are added to the Tier 2 Figure 9.4-1 Control Building HVAC Flow Diagram inlet and exhaust sections to make them consistent with Tier 1 depictions.

The inlet air handling fans and equipment are rearranged and inlet tornado dampers are added to the Tier 2 Figure 9.4-3 Secondary Containment HVAC System diagram to make it consistent with the Tier 1 depiction.

Tornado dampers are added to the inlet and exhaust sections and fire dampers are removed from Tier 2 Figure 9.4-4 R/B Safety Related Electrical Equipment HVAC System to make them consistent with Tier 1 depictions.

In addition, this departure also clarifies that the nonsafety-related system design temperature limits are at the 1% exceedance values provided in Table 2.0-2. This includes the Secondary Containment HVAC System.

The change to the tornado dampers air handling fans and equipment and the fire dampers are a standard departure (STD). The change to the nonsafety-related system design temperature limits are a site specific departure (STP).

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. As stated above, the changes are made to the Tier 2 drawings to make them consistent with the Tier 1 drawings. The changes have no effect on the frequency or consequences of accidents, or the probability or consequences of malfunctions. The systems involved are not relied upon for ex-vessel severe accident mitigation.

As a result, prior NRC approval of this change is not required.

STD DEP 9.4-9, Turbine Building HVAC System

Description

This departure revises the STP 3&4 Turbine Building HVAC design room temperature, system air flow, and main heating coil. The changes incorporated into the FSAR are as follows:

- Subsection 9.4.4.1.2 (1) lists the Turbine Building HVAC areas and specifies room temperature design limits. Some of the area and room temperatures specified in the DCD are unnecessarily conservative and are revised. Specifically:

- Minimum temperature in Turbine Building changed from 15°C to 10°C. (consistent with other HVAC systems described in the ABWR DCD, such as R/B HVAC and C/B HVAC)
- Steam tunnel area maximum temperature changed from 49°C to 60°C. (consistent with Table 3I-6)
- Moisture separator compartment maximum temperature changed from 49°C to 60°C. (consistent with Table 3I-6)

- The Turbine Building HVAC system changed from a recirculating air flow system to a once-through air flow system to minimize contamination/exposure for controlled areas of the Turbine Building. This change is reflected in Section 9.4.4.1.2 (3) & (6), 9.4.4.2.1, 9.4.4.2.1.1, 9.4.4.2.1.2, Figure 9.4-2a, and Figure 9.4-2b (Sheets 1 & 2).
- The Turbine Building HVAC main heating coil changed from a hot water coil to an electric heater coil. This change is reflected in Sections 9.4.4.2.1.1, 9.4.4.2.2.1, Figure 9.4-2a and Figure 9.2-2b (Sheet 1). Also, the last column is deleted from Table 9.4-5c (Steam to Hot Water Heat Exchanger Area).

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII. B.5. There is no impact on any Tier 1 or Tier 2*, Technical Specifications, Basis for Technical Specifications or operation requirements as a result of these changes.

This departure changes STP 3 & 4 Turbine Building HVAC design room temperature, system air flow, and main heating coil. Although this change alters some area and room temperatures specified in the DCD that are unnecessarily conservative, it does not affect any other plant SSCs; therefore, there is no effect on any accident previously evaluated in the DCD or fission product barrier. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, the change has no adverse impact and does not require prior NRC approval.

STD DEP 9.5-1, Diesel Generator Jacket Cooling Water System

Description

The reference ABWR DCD stated that the Diesel Generator Jacket Cooling Water System conformed to the inspection and testing requirements in Regulatory Guide (RG) 1.108. RG 1.108 was withdrawn in August 1993 with the issuance of RG 1.9, Rev. 3, which endorses IEEE-387 and addresses qualification, preoperational and periodic testing of the diesel generators. As a result, references to RG 1.108 are superseded by the requirements of RG 1.9.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. As noted above, this change updates reference to the NRC requirements by deleting reference to an obsolete RG and replacing it with the current RG. The existing diesel generator jacket cooling water system has been evaluated to Regulatory Guide 1.9 and shown to meet these current requirements. There is no change to any design or function of an SSC important to safety. This change has no impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety. There is no change to any design basis for a fission product barrier nor are there any new accident scenarios created. There is no impact on the probability or consequences of an ex-vessel severe accident.

Based on the results of this evaluation, prior NRC approval is not required.

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

Description

The reference ABWR DCD contained specific engineering design details about the fusible plugs based on an older design concept and patent application, but the actual fusible plugs were never built and tested. The changes made to the STP 3 & 4 FSAR describe the fusible plugs in less prescriptive generic terms to the system design requirements and incorporate design experience from actual design and test results:

- Clarified that 260°C is the nominal temperature for fusible plugs to open.
- Specified the opening temperature as 260°C ± 10°C.
- Added clarification of the isolation valve contained in each piping line in the lower drywell which is locked open during normal operation.
- Replace specific design details of the fusible plug configuration with less prescriptive generic functional and operational characteristics.
- Clarified that the fusible plug valves are not ASME Code components.
- Clarified that the temperature of the surrounding air in the drywell is the measurement point for the opening temperature.
- Revised testing information and expanded the requirement to permit the functions of the fusible plugs to be tested separately, if applicable.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This departure does not change any Tier 1 information, Tier 2* information, Technical Specifications, bases for the Technical Specifications, operational requirements, or design, performance, or testing requirements.

This departure includes two basic changes. First, this departure clarifies the performance and test requirements for the fusible plug valves. For example, instead of only specifying a nominal opening temperature for this valve, the change adds a tolerance to that temperature and further specifies how that temperature is measured. Second, the departure removes specific design details which were in the DCD and replaces them with less prescriptive, more generic performance requirements. Those performance requirements are unchanged from the DCD, but this departure provides additional flexibility to satisfy those requirements. As such, this departure does not change any design, performance or testing requirements for these valves, which are required to function following a severe accident. As a result, this departure has no impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety. Furthermore, it does not change the performance of any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect the performance of features for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 9.5-3, System Description - Reactor Internal Pump Motor-Generator Sets

Description

This standard departure consists of several changes to the technical description of the non-safety Motor-Generator (MG) Set equipment that provides power to connected reactor internal pumps (RIPs). These changes are being made to clarify the original DCD technical descriptions or to reflect changes in the actual equipment design implementation details that have evolved since the original DCD descriptions were written. Basic changes in Section 9.5.10.2 are: (1) a clarification in the first paragraph that there is more than one auxiliary transformer; (2) An MG set to ASD RIP loads interface is through three vacuum circuit breakers and three ASD input transformers. In Paragraph 7.7.1.3(7), a specific power device type “gate-turn-off (GTO)” is deleted. The purpose of this sentence is to describe how to implement the Recirculation Pump Trip (RPT) function in the ASDs. The description is consistent with the ASD design. Paragraph 7.7.1.3(8)(c) includes clarification changes.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, Technical Specifications, Bases for Technical Specifications or operational requirements as a result of these changes. The recirculation flow control system (RFC), which includes four adjustable speed drives (ASDs) and two MG Sets and associated ASDs, has no safety setpoints (see DCD Tier 2 Section 7.7.1.3 (13)).

The departure involves the electrical side of the MG-set/ASD design, but not the function. The design change reflects changes in the actual equipment design details that have been implemented in operating plants. The MG-set/ASD equipment, which has a nonsafety-related function of controlling the speed of the RIPs, is for power generation purposes only (see DCD Tier 2, Sections 7.7.1.3 (10) and 9.5.10.3), and evaluated in the Accident Analyses (see DCD Tier 2, Sections 15.2.6 and 15.3.1). In Section 15.2.6, "Loss of Non-Emergency AC Power to Station Auxiliaries", the analysis includes the six RIPs powered by MG sets that are capable of maintaining their original speeds for one second. This COLA change does not affect the analysis. In Section 15.3.1 "Reactor Internal Pump Trip", the analysis considers a loss of three RIPs. The change from one breaker per three ASDs to one breaker per ASD causes a loss of one RIP in a single failure event of a breaker, ASD or RIP, which is bounded by the current analysis. There is no impact on the probability or consequences of a previously evaluated accident. Consequently, there is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety.

The MG-set/ASD equipment have not been identified as a design feature in the DCD for mitigating an ex-vessel severe accident. The likelihood or consequences of a severe accident are not impacted. Therefore, this change has no adverse impact and prior NRC approval is not required.

STD DEP 9.5-4, Lighting and Servicing Power Supply System

Description

Reference ABWR DCD Subsection 9.5.3 provides for the use of mercury lamps (or equivalent) for high ceilings, except where breakage could introduce mercury into the reactor coolant system. This standard departure replaces the mercury lamps with high-pressure sodium (HPS) lamps. All references to mercury lamps have been replaced with HPS lamps. This standard departure is being taken because the Federal Energy Policy Act of 2005 bans the use of mercury vapor ballasts manufactured or imported after January 1, 2008.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, bases for technical specifications or operational requirements as a result of these changes.

Changing the high ceiling lamps from mercury vapor to HPS only affects the lighting design. The removal of mercury vapor precludes the inadvertent introduction of mercury into the reactor coolant system. It has no effect on any safety systems and provides equivalent lighting. Thus this departure does not have any effect on the frequency of occurrence or consequences of accidents or malfunction of SSCs important to safety previously analyzed.

This change involves systems which are not relied upon for mitigation of ex-vessel severe accident. Therefore the likelihood or consequences of an ex-vessel severe accident is not impacted.

As a result of this evaluation, prior NRC approval of the change is not required.

STP DEP 9.5-6, Diesel Generator Fuel Oil Storage and Transfer System

Description

This site-specific departure addresses the following design changes from the reference ABWR DCD:

- The sample connection for the Fuel Oil Storage Tank is relocated slightly above grade elevation. The fill connection is relocated at grade elevation and the vent is extended to an elevation that exceeds the maximum flood level at STP 3 & 4.
- The fuel oil storage tanks are relocated in concrete vaults underground. Stick gauge access and a gravity drain from the bottom of the tank will be added. Piping will be routed underground in concrete tunnels between the storage tanks and the Reactor Building. Cathodic protection is deleted because piping and tanks will not be directly buried.
- Locked closed isolation valves have been added to the fill and sample lines.
- A second transfer pump for the Diesel Generator Fuel Oil System has been added and the pumps have been relocated inside the 7-day storage tank as a result of the STP 3 & 4 flood level.

Evaluation Summary

This departure for design improvement to the Diesel Generator Fuel Oil Storage and Transfer System has been evaluated pursuant to and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This change is considered a necessary design upgrade because of the potential STP 3 & 4 flooding level. It does not adversely affect any functional or safety requirements for storage and transfer but instead is an upgrade as described above. Since this change does not affect any other plant SSCs, there is no effect on any accident previously evaluated in the DCD. This departure does not change any Tier 1 information, Tier 2* information, Technical Specifications, bases for the Technical Specifications, any other underlying design or other operational requirements. Furthermore, it does not change any plant physical features (other than those affected by this design change), SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STP DEP 9.5-7, Fire Protection - House Boiler Area of the Turbine Building

Description

An electrically-heated house boiler will replace the fuel oil-heated boiler. Therefore, fuel oil will not be a part of the combustible loading in that room. Replacing the fuel oil-heated boiler with an electrically-heated boiler represents an improvement from a fire protection standpoint, as it decreases the combustible loading in room 247 and eliminates a potential open flame ignition source in this plant area. The combustible categories “lubricants” and “cables” remain and the combustible loadings will be quantified by a fire hazards analysis.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change.

This departure involves a reduction in combustible material that improves the fire protection posture of the house boiler area. The house boiler is not important to safety and is not addressed by the ABWR DCD as an initiator of an accident or as a potential cause of a malfunction of structures, systems or components that are important to safety. The house boiler does not have a described function to mitigate the consequences of an accident or to mitigate the consequences of a malfunction of structures, systems or components that are important to safety. The reduction of combustible loading in the house boiler area does not increase the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated by the ABWR DCD. The change associated with this departure does not affect fission product barriers and does not change the method of evaluation used to establish design bases or safety analyses. This change does not affect the probability of occurrence of a severe accident as described by the DCD, nor does it increase the consequences of a severe accident.

Based on this evaluation, prior NRC approval of the change is not required.

STP DEP 10.1-1, Turbine Pressure Description

Description

The reference ABWR DCD description of inlet pressure at the turbine main steam valves in Section 10.1 is correct for BWRs when the turbine inlet pressure is controlled by the pressure regulator, such that turbine inlet pressure varies linearly with reactor power level. For the ABWR, reactor dome pressure is controlled by the pressure regulator, and the turbine inlet pressure is determined by the steam line pressure drop. In this case, reactor vessel pressure is linear, while the pressure at the turbine inlet varies as a function of steam flow and steam line pressure drop. At approximately 70% power, the turbine inlet pressure is higher than the pressure at 100% power. Consequently, increase in flow above 100% will not result in a turbine inlet pressure that exceeds the pressure at approximately 70% power. In the STP 3 & 4 FSAR, the description is changed to the following:

The inlet pressure at the turbine main steam valves reflects reactor power, steam line flow and pressure regulator programming, but never exceeds the pressure for which the turbine components and steam lines are designed.

Tier 2 Section 10.1 of the referenced DCD describes the method to limit turbine inlet pressure for a typical BWR where the turbine control valve is regulated based on turbine inlet pressure. As described in DCD Tier 2 Subsection 7.7.1.8, the pressure regulation uses feedback signals from reactor steam dome pressure sensors. Therefore, Section 10.1 was revised to consistently describe the inlet pressure at the turbine main steam valves based on reactor power and reactor steam dome pressure.

Evaluation Summary

There is no impact on any Tier 1 or Tier 2* DCD, Technical Specifications, Bases of Technical Specifications or operational requirements as a result of this change. This departure does not affect any safety function; therefore there is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety, and this departure does not cause an increase in the dose exposure to the public.

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5 and does not require prior NRC approval.

STP DEP 10.1-2, Steam Cycle Diagram

Description

Figure 10.1-1 of the reference ABWR DCD reflects the steam and power conversion system consisting of four condensate pumps, a prescribed number of filters and demineralizers, three feedwater pumps, two high pressure heater drain tanks, a typical multipressure condenser design, and a main turbine with single stage reheat. For STP 3 & 4, four condensate booster pumps, three low pressure heater drain tanks, and separate No.1 feedwater heater drain coolers are added to this system, with three filters and six demineralizers, four reactor feedpumps, four heater drain pumps, one high pressure heater drain tank, and a turbine design with two stages of reheat. These changes are made to improve the overall cycle efficiency, plant reliability, and availability. Figure 10.1-1 is replaced to indicate these features.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

The described systems are nonsafety-related and provide no safety functions. Therefore, there is no impact on the probability or consequences of an accident or malfunction of SCC important to safety, and no effect on the safety analysis. The departure does not impact any Tier 1 or Tier 2* DCD, Technical Specifications, Bases for Technical Specifications or operational requirements. This change has no adverse impact and does not require prior NRC approval.

STP DEP 10.1-3, Rated Heat Balance

Description

Figure 10.1-2 of the reference ABWR DCD heat balance diagram reflects the turbine and steam cycle design as indicated in Figure 10.1-1 of the DCD. This figure is replaced in its entirety due to the changes in Figure 10.1-1 and the new Toshiba turbine design as described in STP 3 & 4 FSAR Section 10.2. Tier 2 Figure 1.1-2 of the reference ABWR DCD is the reactor heat balance. This figure is updated with the slightly increased (~0.06%) feedwater flow. This figure is also updated to reflect STP 3 & 4 changes in the RIP and CRD purge flows. The reactor heat balance is also updated to use ASME Steam Tables (IAPWS-IF97) and three significant figures for pressures.

Tier 2 Figure 5.1-1, Tier 2 Subsection 5.4.5.2, and Tier 2 Table 11.1-6 are also updated for consistency with the heat balance.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1 or Tier 2* DCD information, the Technical Specifications, Bases for Technical Specifications, any underlying design, or other operational requirements.

The proposed changes in this departure affect only performance data of SSCs not important to safety. The feedwater flow and rated main steam flow increase by less than 0.1% will have a negligible impact on the frequency of occurrence or consequences of an accident. The changes have no impact on fission product barriers or ex-vessel severe accidents.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STP DEP 10.1-4, Valves Wide Open Heat Balance

Description

Figure 10.1-3 of the reference ABWR DCD heat balance diagram reflects the turbine and steam cycle design as indicated in Figure 10.1-1 of the DCD for turbine valve wide open conditions. This figure is replaced in its entirety due to the changes in Figure 10.1-1 and the new Toshiba turbine design as described in STP 3 & 4 FSAR Section 10.2. The changes have no impact on safety or transient analysis assumptions. The inlet feedwater temperature and flow remain the same as those in the DCD.

Evaluation Summary

This departure has been evaluated and determined to comply with the requirements in 10 CFR 52, 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.

This departure reflects the different design due to the changes in Figure 10.1-1 and the new Toshiba turbine compared to the reference DCD, and the differences from the plant specific design affect only the performance data for each system or component. All information related to the system and component data described in the heat balance diagram is nonsafety-related.

Therefore, the changes for this departure do not affect the inputs for evaluation for an accident, malfunction of a SSC important to safety, design basis limit for a fission product, the establishing design bases, safety analyses, nor ex-vessel severe accident, in the DCD, and do not create a possibility for an accident of a different type than any evaluated previously in the DCD.This change has no adverse impact and does not require prior NRC approval.

STP DEP 10.2-1, Turbine Design

Description

Compared to the product that formed the basis of the reference ABWR DCD, the following are the significant technical differences in the latest turbine design:

- Two stages of reheat in the steam cycle instead of the single stage utilized in the reference ABWR DCD, to improve turbine steam cycle efficiency
- Replacing the separate reheater shells with symmetrically combined reheater shells of two stages of four U-tube bundles, reducing the number of moisture separator reheaters (MSRs) from four to two.
- Separate Intermediate Stop and Intercept Valves are applied instead of the Combined Intermediate Valves described in the DCD, to provide for enhanced performance, reliability, and maintainability.

Additionally, the following subsections are revised to provide clarification and changes based on the design, procedures, and vendor/manufacturer recommendations:

- Subsection 10.2.2.2-Component Descriptions, including the MSRs, Intermediate Stop and Intercept Valves, Low Pressure Turbines, Extraction Non-Return Valves, and the Generator
- Subsection 10.2.2.6-Turbine Protection System to describe main turbine trip logic as well as trip signals and trip response
- Subsection 10.2.3.5-Preservice Inspection Procedures and Acceptance Criteria
- Subsection 10.2.3.6-Inservice Inspection Requirements for Turbine Generator components and Turbine Steam Valves

Evaluation Summary

The changes do not result in any functional departure from the referenced DCD. The Turbine Main Steam System is classified nonsafety-related. All the changes associated with this departure are to SSCs that are not important to safety, and do not alter the function of SSC important to safety as described in the DCD. Therefore, the changes have no impact on the probability or consequences of an accident or malfunction of SCC important to safety. Therefore these changes do not affect the safety or transient analysis assumptions.

This departure has been evaluated pursuant to with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1 and Tier 2* DCD information, Technical Specifications, Bases for Technical Specifications, or operational requirements as a result of these changes. Therefore, this departure has no adverse impact and does not require prior NRC approval.

STP DEP 10.2-2, Turbine Rotor Design

Description

In sections related to turbine rotor integrity, the reference ABWR DCD considered rotors of built-up construction. Today, the standard is the use of monoblock rotor forgings. Clarification has been provided in the STP 3 & 4 FSAR to enhance the description of turbine overspeed, design speed and their relationship to turbine rotor integrity.

The changes have no impact on safety or transient analysis assumptions. The monoblock rotor design greatly reduces the probability of turbine missiles due to overspeed, improves reliability and reduces maintenance.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VI II.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.

The proposed change provides a turbine rotor design with improved reliability which reduces the probability of turbine missiles during overspeed conditions. As such, the design improvement has no adverse effect on accidents, malfunctions of a SCC important to safety, design basis limits for a fission product, safety analyses, the establishing design bases, safety analyses, and ex-vessel severe accidents, previously evaluated in the DCD, and does not create a possibility for an accident of a different type than any evaluated in the DCD. Therefore, this departure has no adverse impact and does not require prior NRC approval.

STP DEP 10.2-3, Turbine Digital Control

Description

Several modifications to the control logic for the turbine generator are described in Section 10.2. These modifications resulted from implementation of digital turbine controls for machine protection and reliability. This departure implements the following modifications:

- The control system uses electronic monitoring for control and overspeed protection of the main turbine.
- Redundancy for overspeed trip is implemented using electrical overspeed trip devices based on a hardware configuration. The overspeed trip system consists of the Primary and Emergency overspeed trip functions with two-out-of-three logic employed in each trip circuitry for additional reliability.

The expected speed range resulting from sudden loss load as 106 to 109% and the limit of turbine speed when overspeed trip devices activate as 120% were defined.

In addition, Subsection 10.2.2.7 is modified to define the frequency to which main turbine valves (stop valves, control valves, intermediate stop valves, and intercept valves) are exercised to include verification of the fast closure function.

Evaluation Summary

These modifications allow for full online testability of any protective function and significantly reduce the possibility of tripping the main turbine during testing. Most major components of the overspeed monitoring and control system are located in low radiation areas and are designed for safe, online troubleshooting and maintenance of mission critical components (e.g. turbine trip logic circuit and turbine valve control function).

Reliability for the electrical trip system is achieved by using two sets of redundant speed sensing probes, which input to the independent Primary and Emergency Trip hardware logic in the control system. A common cause failure of the software-based logic cannot occur because the trip logic is based on a hardware configuration.

Also this departure defines the frequencies to exercise main turbine valves, which will provide a basis for improved component reliability.

The main turbine and turbine control system are classified as nonsafety-related. The turbine digital controller increases plant availability because a single failure will not result in a turbine trip and plant shutdown. There is no effect on the frequency or consequences of any accidents or malfunctions of SSC important to safety previously evaluated in the ABWR DCD. The overspeed protection system is not identified as equipment needed for any fission product barrier or mitigation of ex-vessel severe accidents. Therefore, this change has no impact on the probability of an ex-vessel severe accident, there is no possibility of a new type of accident, and there is no impact on fission product barriers or ex-vessel severe accident events.

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on Tier 1 or Tier 2* DCD, Technical Specifications, Bases of Technical Specifications, or operational requirements as a result of the changes. Therefore, this departure has no adverse impact on the safety analysis and does not require prior NRC approval.

STP DEP 10.2-4, Bulk Hydrogen Storage

Description

Subsection 10.2.2.2 of the reference ABWR DCD states that bulk hydrogen for the generator is stored outside but near the turbine building. This departure changes the description to state that bulk hydrogen for STP 3 & 4 will be stored well away from the power block buildings. Storing the bulk hydrogen away from the power block buildings reduces the probability of inadvertent explosion or fire causing damage to the buildings.

Evaluation Summary

This departure has been evaluated in accordance with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This change does not adversely affect the frequency of occurrence or impact of an accident on any SSCs, since any damage to the Power Block by inadvertent explosion, fire or missile generated due to bulk hydrogen tank failure will be reduced by increasing the distance between the Bulk Hydrogen Storage Building and the Power Block.

In addition, this departure only moves the location of the Bulk Hydrogen Storage System and does not make any other changes. Therefore, this change does not result in new potential accident scenarios or new analysis methodologies.

Thus, the departure has no adverse impact, and prior NRC approval is not required.

STD DEP 10.3-1, Main Steam Line Drains

Description

Subsection 10.3.2.1 of the reference ABWR DCD states that the drains from the steamlines inside containment are connected to the steamlines outside the containment to permit equalizing pressure across the MSIVs during startup and following a steamline isolation. FSAR Subsection 10.3.2.1 expands that discussion to state that the Main Steam System also serves as the main steam line leakage path to contain the radioactive steam which passes the main steam isolation valves before they close to isolate the reactor under emergency conditions. The discussion provides the details of design that provide the leakage path. Details concerning the slope of the main steam line drain piping are also added.

Evaluation Summary

This departure updates the Tier 2 Section 10.3.2.1 description of the main steam supply to include a discussion of the function of the main steam piping as the main steam line leakage path to contain the radioactive steam which passes the main steam isolation valves before they close during emergency conditions. This description is consistent with the design of the piping as provided in DCD Tier 2 Section 3.2.5.3 and does not reflect any change to that design.

Section 10.3.2.1 is updated to include additional details concerning the slope of the steam line drains. This discussion is consistent with the DCD.

These updates to Section 10.3.2.1 are intended to provide further clarification and are consistent with the main steam line and drain piping function and design as provided in the DCD. Consequently, this departure has no effect on any SSC design or function and has no impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analysis. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or function of SSC important to safety, the change does not create a different ex-vessel accident scenario.

This departure does not change any Tier 1 or Tier 2* DCD information, the Technical Specifications, Bases of Technical Specifications, any underlying design, or operational requirements.

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The change has no adverse impact and does not require prior NRC approval.

STD DEP 10.4-1, Turbine Gland Seal Steam

Description

A nonsafety-related gland seal evaporator (GSE) is added to the reference ABWR DCD Turbine Gland Steam System to supply sealing steam to the main turbine shaft seal glands and various turbine valve stems, including the turbine bypass and main turbine stop-control valve stems.

Clean condensate makeup water is supplied to the GSE, which can be heated by either main steam or steam from the cross-around piping. The GSE will provide isolation from the potentially contaminated heating steam and the clean steam supplied to the gland seal system. The gland seal steam that can be supplied from the electrically heated auxiliary steam system is unaffected by the addition of the GSE.

The addition of the GSE will allow operational flexibility and minimize the use of the auxiliary boiler during plant startup and shutdown. Furthermore, the gland seal steam is condensed in the gland seal condenser and the non-condensible gases are discharged to the environment. The use of the clean steam for gland sealing will minimize dose release to the environment and ALARA concerns.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, 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.

As discussed in the departure description, the proposed design provides operational flexibility and minimization of components and dose release to the environment. In addition, the TGSS is a nonsafety-related system. Therefore, the changes for this departure do not affect the inputs for evaluation of an accident, malfunction of a SSC important to safety, design basis limit for a fission product, the establishing of design bases, safety analyses, or evaluation of an ex-vessel severe accident, in the DCD, and do not create a possibility for an accident of a different type than any evaluated previously in the DCD. Consequently, the change has no adverse impact and does not require prior NRC approval.

STP DEP 10.4-2, Main Condenser

Description

The main condenser design described in the reference ABWR DCD utilizes three independent multi-pressure single-pass shells, with each shell containing at least two tube bundles, and series circulating water flow. STP 3 & 4 will utilize three condenser shells cross-connected to equalize the pressure, with each shell containing four tube bundles, and parallel circulating water flow.

This site-specific departure will provide four 25% capacity circulating water pumps discharging into a common header. This departure will also provide the water filling procedure and instrumentation not specifically included in the reference ABWR DCD.

In addition, this departure eliminates the warm water recirculation operating mode to mitigate ice effects, along with deleting the associated warm water recirculation components.

Finally, the Condenser and Circulating water system design data are provided as site specific values.

Evaluation summary

The main condenser is of a conventional design, with the pressure in the three shells equalized. This site-specific departure provides four 25% capacity circulating water pumps discharging into a common header. This provides greater circulating water flow improving the ability of the equalized pressure shell condenser to maintain a higher vacuum, which enhances plant performance.

The warm water recirculation operating mode is eliminated based on historical water temperature data and the results of the potential ice effect evaluation as clarified in Subsection 10.4.5.5 of the FSAR.

Condenser design data in Table 10.4-1 are calculated for the optimized site specific condition of the condenser shell pressure and circulating water flow and temperature in Table 10.4-3.

The changes to the main condenser and Circulating Water System optimize the design for the site and do not alter the design functions specified in the DCD. There is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety.

The procedure and instrumentation for initial water filling for both startup and plant power operation are described in Subsection 10.4.5.2.1, Subsection 10.4.5.2.3 and Subsection 10.4.5.5 of the FSAR. This procedure is used to develop and maintain venting of the system and prevents water pressure surges from damaging the piping or the condenser.

The changes on the Circulating Water System and Main Condenser were made to meet the the site-specific condition and do not alter the design functions specified in the DCD. Therefore, there is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety.

This departure has been evaluated and pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The changes have no adverse impact. As discussed above, the changes do not alter the design basis as described in the DCD. Therefore, 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 and prior NRC approval is not required.

STP DEP 10.4-3, Main Condenser Evacuation System

Description

This site-specific departure adds an additional mechanical vacuum pump, so the design now consists of two vacuum pumps, and changes the source of motive steam supplying the steam jet air ejectors during power operation.

The mechanical vacuum pump system establishes an initial vacuum in the condenser during the initial phase of startup. The vacuum pump may also be put into service when the desired rate of air and gas removal exceeds the capacity of the Steam Jet Air Ejectors. Only one mechanical vacuum pump is required for operation. The additional mechanical vacuum pump is added to serve as a backup instead of Steam Jet Air Ejectors driven by auxiliary boiler steam, to ensure MCES redundancy during startup. The second vacuum pump will enhance reliability during power operation and increased flow capacity during startup will reduce time to achieve required condenser vacuum. This will reduce the time to draw condenser vacuum, thus reduce startup time and enhance secondary system operation.

The site-specific design uses main steam as the main source to drive the Steam Jet Air Ejectors instead of utilizing cross-around steam with main steam as a backup. This eliminates possible transient effects, such as partial loss of condenser vacuum or inadequate steam dilution of radiolytically generated hydrogen, which might occur during a switchover from cross-around steam to main steam.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, 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. The MCES is a nonsafety-related system. Therefore, the changes for this departure do not affect the inputs for evaluation for an accident, malfunction of a SSC important to safety, design basis limit for a fission product, the establishing design bases, safety analyses, nor ex-vessel severe accident, in the DCD, and do not create a possibility for an accident of a different type than any evaluated previously in the DCD. Consequently, the change has no adverse impact and prior NRC approval is not required.

STD DEP 10.4-6, Load Rejection Capability

Description

Because the ABWR standard design has a turbine bypass system capacity of 33% of nuclear boiler rated flow, it can accommodate a 33% load reduction without reactor trip by full opening of the bypass valves. It can also accommodate a turbine trip from 33% power or below without reactor trip. Turbine trip or generator load rejection from power levels above 33% will result in a reactor trip, with attendant opening of SRVs if the trip is from sufficiently high power levels. Subsections 10.4.4.1.2 and 10.4.4.2.1 of the FSAR have been revised accordingly.

This departure also clarifies the description of the Automatic Power Regulator (APR) system and its relation to the turbine bypass valves in COLA Tier 2 Subsection 10.4.4.2.3.

Evaluation Summary

The described change is limited to the step-load reduction capacity without reactor trip, and with operation of the bypass valves.

Signal interfaces between the bypass valves and the Automatic Power Regulator (APR) system are provided as described in Tier 2 Section 7.7. The change does not affect the function of the bypass valves or coordination of the Turbine Bypass System (TBS) controls by the APR system.

Turbine bypass capacity is not changed from the specified input parameter for transient analysis system response in Table 15.0-1. The change is consistent with the function of the bypass valves to open upon turbine trip or generator load rejection and the design analysis results for transients such as load rejection with failure of all bypass valves and feedwater controller failure, maximum demand. The change does not affect any control system associated with bypass valves, and does not change the capacity of the bypass system or its response to a transient. Therefore, the change cannot affect the probability or consequences of an accident or a malfunction of an SSC important to safety previously evaluated in the ABWR DCD, and does not result in an accident or a malfunction of an SSC important to safety other than was previously evaluated. The departure does not affect the assumptions in the analyses for generator load reject or turbine trip with or without the bypass valves, and does not affect the analysis for opening of one or more bypass valves.

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 this change.

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. The change has no adverse impact and prior NRC approval is not required.

STD DEP 10.4-7, Turbine Bypass Hydraulic Control

Description

Tier 2 Figure 10.4-9 is revised to indicate the use of valve position transmitters, one hydraulic accumulator for each bypass valve, the addition of the fast-acting solenoid valve, and the interface with the Steam Bypass and Pressure Control System (SB&PCS) for positioning of the bypass valves. This revision resolves inconsistencies between the figure and text descriptions as evaluated below.

Evaluation Summary

Valve position transmitters are used instead of valve position switches. The triplicated position transmitters are highly reliable, meeting the requirement that no single failure will result in loss of function, and provide signals for fully opened and closed position indications.

The fast-acting solenoid valves allow rapid opening of the bypass valves to mitigate the increase in reactor pressure in the event of a turbine trip or generator load reject, and for testing of bypass valves. The solenoid valve is added to make the figure consistent with the description in Tier 2 Section 10.4.4.2.2.

Signal interfaces from the valve position transmitters to SB&PCS controllers to the fast-acting solenoid valves and servo valves that control the positions of the bypass valves are provided as described in Tier 2 Section 7.7.1.8. The changes do not affect the interface between the Turbine Bypass System (TBS) and SB&PCS and operation of the systems as described in Tier 2 Section 7.7.1.8.

One hydraulic accumulator for each bypass valve is indicated, according to the description in Tier 2 Section 10.4.4.2.2.

Valve position transmitters, fast-acting solenoid valves, hydraulic accumulators, and the bypass valves and controllers are components of TBS and SB&PCS and therefore are nonsafety-related. These changes do not affect the function of the Turbine Bypass System (TBS) to open upon turbine trip or generator load rejection, do not affect the function of the TBS during design basis feedwater controller failure, maximum demand event, as described in DCD Tier 2, Section 15.1.2, and the Technical Specification Bases, and are consistent with the triplicated fault-tolerant design used in Tier 2 Chapter 15 analyses. The changes do not affect any safety function; therefore there is no impact on the probability or consequences of an accident or malfunction of an SSC important to safety, and this departure cannot cause an increase in the dose to the public.

This departure has been evaluated pursuant to with the requirements in 10 CFR 52, 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 this change. These changes have no adverse impact and do not require prior NRC approval.

STD DEP 11.2-1, Liquid Radwaste Process Equipment

Description

The departure includes the use of mobile technology and deletes the forced-circulation concentrator system and other permanently installed liquid radwaste processing equipment. The following liquid waste management system (LWMS) description reflects the changes to the system that have been generated by this standard departure.

The LWMS is composed of three subsystems designed to collect, treat, and recycle or discharge different categories of waste water. The three subsystems are the Low Conductivity (LCW) Subsystem, High Conductivity (HCW) Subsystem, and Hot Shower and Detergent Waste (HSD) Subsystem. The Chemical Drains (CD) are also collected and processed through the HSD Subsystem.

The LCW subsystem collects and processes clean radwaste (i.e., water of relatively low conductivity). Equipment drains and backwash transfer water are typical of wastes found in this subsystem. In the reference ABWR DCD, LWMS processes LCW as follows: The wastes are collected, filtered for removal of insolubles, demineralized on a mixed resin, deep-bed demineralizers for removal of solubles, processed through a second polishing demineralizer, and then routed to condensate storage unless high conductivity requires recycling for further treatment. A second LCW filter, arranged in parallel with the first, is also provided. The LWMS, using the mobile technology introduced with the change, processes LCW as follows: The waste is collected, mixed, sampled, and analyzed and then processed through modular processing units as required to treat the waste. After treatment, the waste is collected in sample tanks, sampled and analyzed, and recycled to the condensate storage tank, reprocessed if required, or discharged.

The HCW subsystem collects and processes dirty radwaste (i.e., water of relatively high conductivity and solids content). Floor drains are typical of wastes found in this subsystem. In the ABWR DCD LWMS processes HCW as follows. The wastes are collected, chemically adjusted to a suitable pH for evaporation, and concentrated in a forced-circulation concentrator with a submerged, steam-heated element to reduce the volume of water containing contaminants and to decontaminate the distillate. The distillate is demineralized to remove any soluble contaminants that could potentially be carried over from the concentrator. The LWMS, using the mobile technology introduced with the change, processes HCW as follows. The waste is collected, mixed, sampled, and analyzed and then processed through modular processing units as required to treat the waste. After treatment, the waste is collected in sample tanks, sampled and analyzed, and reprocessed if required, or discharged.

The ABWR DCD LWMS utilizes submerged-feed, forced circulation concentrators. Chemical addition and sampling equipment are provided for feed pretreatment to prevent excessive fouling and subsequent high carryover, and to protect the concentrator from corrosion. Concentrator feeds are concentrated to the required specific gravity and discharged to the solids handling equipment.

This equipment is replaced with a mobile liquid radwaste processing system consisting of modular liquid radwaste processing units that utilize appropriate technology, which is available, that will perform satisfactorily and produce an acceptable quality of treated waste. The processes currently envisioned include filtration, reverse osmosis, and ion exchange. The liquid radwaste processing system components are in modules that are designed for ease of installation and replacement due to component failure and/or technology upgrade. The waste from the LWMS process units is discharged to the Solid Waste Management System.

Other changes to the LWMS include changes to the number and capacity of the tanks and pumps, which will be permanently installed. The system is sized to process the normal liquid radwaste flows within a four-hour shift, five days per week, and one day of the maximum daily flow and four days of the normal daily flow within an eight-hour shift, five days per week.

The ABWR DCD HSD subsystem collects and processes detergent wastes from personnel showers and laundry operations. Normally, hot shower, detergent wastes, and storm water are collected in the detergent tank and processed through a detergent filter and discharged.

The HSD subsystem is changed to collect wastes from personnel showers and laundry operations, process the waste through a filter, and collect the processed waste in a sample tank. The processed waste is sampled and analyzed and discharged or processed through the HSD subsystem if necessary. The storm water is not managed using the LWMS.

Evaluation Summary

This departure has been evaluated pursuant the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on Tier 1, Tier 2*, technical specifications or operational requirements as a result of this change.

The functional requirements of the LWMS, which is not shared between STP 3 & 4, are not changed by this departure except the storm water is not processed using the LWMS. The system is consistent with the description contained in the reference ABWR DCD except that the permanent liquid waste processing components are replaced with mobile liquid waste processing modules. In addition, the number and capacity of the permanently installed tanks and pumps is changed.

No fundamentally new processes or equipment are introduced by the changes to the LWMS and the complexity of the system is reduced (i.e., the permanent filters, forced recirculation evaporator, and ion exchangers are removed and replaced with mobile filters, reverse osmosis units, and ion exchangers). The reverse osmosis units are essentially very fine membrane filtration units and therefore are not a fundamentally new process.

The limiting accident for the LWMS is the failure of the LCW Collection tank and the subsequent airborne release, which is described in Section 15.7.3 of the DCD. The capacity of each LCW tank is reduced (from 430 m^3 to 140 m^3) and the number increased (from two to four). The design standards for the LCW tanks are not changed. Therefore, the proposed change does not result in more than a minimal increase in the frequency of the limiting accident previously evaluated in the DCD.

A complex component, which is more prone to malfunction than other components in the LWMS (i.e., forced recirculation evaporator), is removed as part of these changes. Other changes use components that are comparable to those described in the DCD.

As described in Section 15.7.3.1 of the DCD, a liquid radwaste release caused by operator error is considered a remote possibility. The administrative and physical controls for the release system are not changed. Therefore, the proposed change does not result in more than a minimal increase in the occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the DCD.

The limiting accident associated with the LWMS is the failure of the LCW tank and the subsequent airborne release which is described in Section 15.7.3 of the DCD. The volume of the LCW tank is reduced. Therefore, the radionuclide inventory in the tank is reduced, thereby reducing the potential consequences of the accident. The description of the accident in DCD Section 15.7.3 states that the tank is located in a Seismic Category I Structure. As part of these changes, the Radwaste Building structure will be designed in accordance with the seismic requirements of Regulatory Guide 1.143 and will not be Seismic Category I. However, the tank cubicles will be lined with steel to a height capable of retaining the contents of the tank. Therefore, release to the groundwater is not considered credible. Therefore, the proposed change does not result in more than a minimal increase in the consequences of the limiting accident previously evaluated in the DCD.

A component which contains concentrated radionuclides at high temperatures (i.e., forced recirculation evaporator) is removed as part of these changes. Other changes use components that are comparable to the design described in the DCD. As described in Section 15.7.3.1 of the DCD, a liquid radwaste release caused by operator error is considered a remote possibility. The proposed change reduces the capacity of the sample tanks that would be emptied during an inadvertent release. The administrative and physical controls for the release system are not changed. This will reduce the consequences of an inadvertent release. Therefore, the proposed change does not result in more than a minimal increase in the consequences of the malfunction of a SSC important to safety.

No fundamentally new processes or equipment are being introduced by the changes to the LWMS. Therefore, the proposed change does not create the possibility for an accident of a different type than evaluated previously in the DCD.

No fundamentally new processes or equipment are being introduced by the changes to the LWMS. Therefore, the proposed changes do not create the possibility for a malfunction of an SSC important to safety with a different result than evaluated previously in the DCD.

The changes to the LWMS do not involve any interaction with the fuel, reactor system boundary, or the containment boundary. Therefore, the proposed change does not affect the fission product barrier as described in the DCD.

The LWMS design basis waste quantities and characteristics are the same (except for a lower maximum flow of storm water to a sample tank). The LWMS safety analysis is described in Section 15.7.3 of the DCD and the method of performing the analysis does not change. Therefore, the proposed change does not result in a departure from the method of evaluation described in the DCD used in establishing the design basis or in safety analysis.

The changes to the LWMS do not involve any interaction with fuel, reactor system boundary, or the containment structure or interact directly with systems associated with ex-vessel severe accidents. Therefore, there is no substantial increase in the probability of an ex-vessel severe accident such that a particular ex-vessel severe service accident previously reviewed and determined to be not credible could become credible.

The changes to the LWMS do not involve any interaction with fuel, reactor system boundary, or the containment structure or interact directly with systems associated with ex-vessel severe accidents or severe accident mitigation. Therefore, there is no substantial increase in the consequences to the public of a particular ex-vessel severe accident that was previously reviewed.

Based on this evaluation, prior NRC approval of the change is not required.

STD DEP 11.3-1, Gaseous Waste Management System

Description

This departure makes the following changes to the gaseous waste management system:

- Changes the offgas recombiner from an integral unit to independent pre-heater, recombiner, and condenser arranged in a recombiner train.
- Adds an offgas evacuation system downstream of the HEPA filter to stabilize the offgas flow to the plant exhaust.
- Revises the charcoal adsorber vault temperature to a tighter range to maximize charcoal efficiency.
- Changes the number of charcoal adsorber vessels from nine (1 guard bed and 8 adsorbers) to five (1 guard bed and 4 adsorbers). Also, the arrangement of the charcoal adsorbers is changed from four parallel lines, each with two adsorbers in series, to four bigger adsorbers in series.
- Revises the mass of charcoal in each of the charcoal adsorber vessels from 13,600 kg (for the 8 adsorbers) to 27,200 kg (for the 4 bigger adsorbers). The total mass of charcoal in the adsorbers is unchanged. Note that the accident analyses in Section 15.7 assumes bypass of the charcoal adsorbers downstream of the guard bed, so the accident analyses are unaffected.
- Changes the mass of charcoal in the guard bed from 4,500 kg to 4,721 kg in Section 11.3 to be consistent with the accident analysis described in Section 15.7.1.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, Technical Specifications, Bases for Technical Specifications or operational requirements as a result of these changes. The change from an integral recombiner to a recombiner train provides the same function as the integral recombiner and has proven operational experience. The addition of the evacuation system to assure stable offgas flow to the plant exhaust is an enhancement to the offgas discharge and does not change the offgas function. Restricting the charcoal adsorber vault temperature optimizes the charcoal performance and does not change the design basis. The change in the number of charcoal adsorbers, and the change in adsorber configuration and vessel size, provides the same function as the DCD design and has proven operational experience.

This departure does not change any accident evaluation including those in ABWR DCD Tier 2 Subsection 15.7. There is also no impact on the probability or consequences of an accident or malfunction of an SSC important to safety. Furthermore, there is no impact on fission product barriers. Therefore, this change has no adverse impacts and does not require prior NRC approval.

STD DEP 11.4-1, Radioactive Solid Waste Update

Description

Described below are Solid Waste Management System (SWMS) modifications addressed in this standard departure:

The solidification system, the dryer system, and the incinerator system are deleted because equipment operation and maintenance difficulties negatively impact the effectiveness of these processes. The compactor system is deleted as well.

A second spent resin storage tank is added to provide the capability to keep the spent resins from the Condensate Purification System and the spent resins from the LWMS mobile systems ion exchangers in separate spent resin storage tanks for radioactive decay and storage. This segregation allows the used condensate polishing resin from the Condensate Purification System may be used in the HCW demineralizer in the high conductivity waste subsystem. The reuse of the condensate resin helps to minimize the generation of radioactive waste.

A Liquid Waste Backwash Receiving Tank is added to collect the backwash from the Liquid Waste Management System (LWMS) mobile process units for transfer to the Phase Separators.

The SWMS mobile system consists of equipment modules, complete with all subcomponents, piping and instrumentation and controls necessary to operate the subsystem. Solid wet radwaste processing is performed using mobile dewatering processing subsystem. The mobile dewatering processing subsystem is comprised of dewatering fillhead assembly, dewatering pump skid, valves, control console and dewatering container. The mobile dewatering processing subsystem includes the adequate shielding required between the radiation sources of the modules and access and service areas in the radwaste building. The SWMS mobile system components are in module(s) that are designed for ease of installation and replacement due to component failure and/or technology upgrade.

Evaluation Summary

This departure has been evaluated pursuant the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on Tier 1, Tier 2*, technical specifications or operational requirements as a result of this change.

The functional description of the SWMS, which is not shared between STP 3 & 4, is not significantly changed by this departure. The system is consistent with the description contained in the reference ABWR DCD except that the permanent volume reduction (incinerator and dryer), solidification, and dry active waste compaction units are replaced with a mobile solid waste dewatering unit. The specific processes will be selected prior to initial plant operation and may be changed as appropriate. In addition, the number and capacity of the permanently installed tanks and pumps is changed.

No fundamentally new processes or equipment are introduced by the changes to the SWMS and the complexity of the system is reduced (i.e., the incinerator, the dryer, the compactor, and the radwaste solidification system are removed). The limiting accident for the Radwaste Building is the failure of the Low Conductivity Waste (LCW) collector tank and the subsequent airborne release which is described in Section 15.7.3 of the DCD. The capacity of the SWMS tanks is not increased, therefore, the LWC Collector Tank failure remains the limiting accident. Therefore, the proposed change does not result in more than a minimal increase in the frequency of the limiting accident previously evaluated in the DCD.

No fundamentally different processes or equipment are introduced by the changes to the SWMS. Complex components, which are more prone to malfunction than other components in the SWMS (i.e., incinerator, the dryer, the compactor, and the radwaste solidification system), are removed as part of these changes. Other changes use components that are comparable to those described in the DCD. Therefore, the proposed change does not result in more than a minimal increase in the occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the DCD.

The description of the limiting accident associated with the Radwaste Building, which is described in DCD Section 15.7.3, states that the Radwaste Building is a Seismic Category I Structure. As part of these changes, the Radwaste Building structure will be designed in accordance with the seismic requirements of Regulatory Guide 1.143 and will not be Seismic Category I. However, the tank cubicles are lined with steel to a height capable of retaining the contents of the tank. Therefore, postulated release to the groundwater is not considered credible. Therefore, the proposed change does not result in more than a minimal increase in the consequences of the limiting accident previously evaluated in the DCD.

No fundamentally different processes or equipment are being introduced by the changes to the SWMS. Components, which contain concentrated radionuclides at high temperatures (i.e., the incinerator, the Concentrated Waste Tank, and the dryer), are removed as part of these changes. Other changes use components that are comparable to the design described in the DCD. Removal of these systems reduces the potential consequences of a malfunction by eliminating the potential for malfunctions. Therefore, the proposed change does not result in more than a minimal increase in the consequences of the malfunction of a SSC important to safety.

No fundamentally new processes or equipment are being introduced by the changes to the SWMS. Therefore, the proposed change does not create the possibility for an accident of a different type than evaluated previously in the DCD.

No fundamentally new processes or equipment are being introduced by the changes to the SWMS. Therefore, the proposed changes do not create the possibility for a malfunction of an SSC important to safety with a different result than evaluated previously in the DCD.

The changes to the SWMS do not involve any interaction with the fuel, reactor system boundary, or the containment boundary. Therefore, the proposed change does not affect the fission product barrier as described in the DCD.

The SWMS design basis waste quantities and characteristics are similar to those described in the DCD. The limiting safety analysis for the Radwaste Building is described in Section 15.7.3 of the DCD and the method of performing the analysis does not change. Therefore, the proposed change does not result in a departure from the method of evaluation described in the DCD used in establishing the design basis or in safety analysis.

The changes to the SWMS do not involve any interaction with fuel, reactor system boundary, or the containment structure or interact directly with systems associated with ex-vessel severe accidents. Therefore, there is no substantial increase in the probability of an ex-vessel severe accident such that a particular ex-vessel severe service accident previously reviewed and determined to be not credible could become credible.

The changes to the SWMS do not involve any interaction with fuel, reactor system boundary, or the containment structure or interact directly with systems associated with ex-vessel severe accidents or severe accident mitigation. Therefore, there is no substantial increase in the consequences to the public of a particular ex-vessel severe accident that was previously reviewed.

Based on this evaluation, prior NRC approval of the change is not required.

STP DEP 11.5-1, Process and Effluent Radiation Monitoring and Sampling System

Description

There are several changes that have been made for this system:

- Functional Requirements set forth in the reference ABWR DCD will be met, but implementation of design and specific equipment is vendor-based.
- References to specific detector types, such as digital gamma sensitive Geiger-Mueller, ionization chamber, or scintillation detector, were deleted. Specific detector types will be selected later in the project based on state of the art and availability.
- Trip functionality for radiation monitors has been modified such that downscale (low) and inoperative are combined into one trip circuit rather than two separate circuits because these two trips are used for the common purpose of detecting equipment failures. Thus, each radiation monitor has three trip circuits: two upscale and one downscale/inoperative. Each trip is determined by the radiation monitor and then sent to the main control room for visual display.
- As for radiation unit to express the range of radiation monitor, Sievert is preferred to using Gray. Sieverts specifically addresses absorbed radiation dose in human tissue while Gray refers to radiation dosage in any material.
- Recorders have been removed because data recording is performed by trending software in the Digital Control and Instrumentation System.
- STP 3 & 4 will not have an incinerator for burning low-level radwaste, so the incinerator stack discharge radiation monitor is not required. Sections and references to this monitor have been removed.
- References to specific calibration techniques and maintenance procedures are removed. These techniques and methods, such as calibration reproducibility, error, precision, and timelines for maintenance, are specific to site procedures or are supplied by the equipment vendors.
- FSAR Table 11.5-1 provides estimated channel ranges. Channel ranges will only be finalized after analyses and calculations are complete.
- Warning alarms are provided in the text of the specific section for each radiation monitor and do not need to be provided in the table. Table 11.5-2 and Table 11-5- 3 provide expected activity, dynamic detection ranges and sensitivity. Dynamic detection ranges are calculated based on the radionuclides and the sensitivity of the radiation monitor. As the sensitivities are vendor provided, the dynamic detection range is estimated. Sensitivities are not included in the table as they are vendor provided.
- The bypass valve closure trip of the Offgas Post-Treatment Radiation Monitor is corrected to High-High to be consistent with DCD IED (Figure 7.6-5).

- The High-High alarm for the Gland Seal Condenser Exhaust is added to be consistent with DCD (Figure 7.6-5).

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1 and Tier 2* technical specifications, basis for technical specifications and operational requirements as a result of these changes.

Implementation of detailed design and specific equipment is changed. But Functional requirements set forth in the reference ABWR DCD are not changed.

The Radiation Unit is changed to Sievert. Sievert specifically addresses absorbed radiation dose in human tissue while Gray refers to radiation dosage in any materials.

The recorders are removed because data recording is performed by trending software in the Digital Control and Instrumentation System. There will not be an incineration burning low-level radwaste, so the incineration stack discharge radiation monitor is not required.

The trip descriptions for Offgas Post-Treatment Radiation Monitor and the alarm descriptions for Gland Seal Condenser Exhaust Radiation Monitor are corrected to be consistent with DCD IEDs. Consequently, there is no impact on the probability or consequences of an accident, malfunction of an SSC important to safety, or the likelihood or consequences of a severe accident.

Based on this evaluation, prior NCR approval of these changes is not required.

STD DEP 12.3-1, Cobalt Content in Stainless Steel

Description

This departure revises the requirements for the material specification for the stainless steel component exposed to reactor coolant with specific reference to the cobalt content in the stainless materials.

The vendors supplying the material cannot reasonably achieve the cobalt limits in all cases. A graded approach to cobalt concentrations has been taken by using various grades of low cobalt stainless steel, with the material in the core receiving the least amount of cobalt. The cobalt concentrations are allowed to increase with the distance from the core. The overall cobalt limit for all reactor vessel material is 0.05 wt percent. Lower target values (aim limits) are provided to the material vendor as goals to trend for.

During the ABWR Certification process the average annual occupational exposure calculation was performed. The reduced cobalt loadings were not considered in that estimate. Therefore, based upon the method used and the assumptions made to evaluate the occupational exposure, materials procured with a 0.05 wt percent maximum cobalt requirement, with lower ALARA target values of cobalt for radiologically significant areas, will have no adverse affect on the estimated occupational exposure.

Evaluation Summary

The departure to clarify the the procurement and use of various grades of low cobalt stainless steel on a graded approach wherein the lowest cobalt material is used in the most radiologically significant areas with increasing cobalt material in less sensitive areas has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There are no significant radiological consequences since the ABWR certification calculation of the average annual occupational exposure did not assume the reduced cobalt loadings, and the results show no adverse effect. This departure does not change the Technical Specifications, any underlying design or other operational requirements. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 12.3-2, Deletion of CUW Backwash Tank Vent Charcoal Filter

Description

This departure deletes the statement in Subsection 12.3.1.4.1 which states that the vent off the CUW backwash tank is fitted with a charcoal filter canister to reduce the emission of radioiodines into the plant atmosphere. A review of the system diagrams for the CUW system show no such filter as part of the approved design. The current design intent is for the CUW backwash tank to be vented into the Reactor Building HVAC System exhaust, which eventually exits the plant via the plant stack as a monitored release.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5.

This departure does not change any Tier 1, Tier 2* information, the Technical Specifications, any underlying design or other operational requirements.

This departure corrects the text description of the backwash tank vent system by deleting reference to a charcoal filter on that vent system which does not exist in the design. As noted earlier, the current design intent is for the CUW backwash tank to be vented into the Reactor Building HVAC System exhaust, which eventually exits the plant via the plant stack as a monitored release. Since this change does not affect any plant SSC, there is no affect on any accident previously evaluated in the plant-specific DCD. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possiblility for an accident of a different type is not created. Also it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 12.3-3, Steam Tunnel Blowout Panels

Description

This departure removes the discussion in Subsections 3.8.4, 12.3.1.4.4 and 12.3.2.3 concerning blowout panels and relief and release pathways associated with the steam tunnel because these statements are inaccurate and conflict with the correct description in revised Subsection 3.8.4 and Subsection 3.12.1.3, where it more appropriately belongs. The design description of the routing and functioning of the blowout panels in Subsections 12.3.1.4.4 and 12.3.2.3 is inaccurate and is not needed in these subsections. This departure also adds the phrase "or equivalent" to the last sentence in Subsection 12.3.1.4.4 describing the use of lead-loaded silicone foam for sealing penetrations.

Evalutation Summary

This departure has been evaluated pursuant to and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure which removes the discussion concerning blowout panels and relief and release pathways associated with the steam tunnel would appear to eliminate a steam flow path; however, because there is no high energy line break (HELB) identified in the room identified in the deleted text (subcompartment SA2 - RHR Pump and Heat Exchanger Room), there is no need for a relief path from this compartment. Additionally, SSCs in the flow paths and adjoining areas are qualified or protected as needed from increased temperature, pressure and jet impingement forces. The design description of the location, routing and functioning of the blowout panels in Subsections 12.3.1.4.4 and 12.3.2.3 is inaccurate and is not needed in those subsections. The correct description of the blowout panels and relief and release pathways associated with the steam tunnel is contained in revised Subsection 3.8.4 and Subsection 3.12.1.3, where it more appropriately belongs. This departure also adds the phrase "or equivalent" to the last sentence in Subsection 12.3.1.4.4 describing the use of lead-loaded silicone foam for sealing penetrations to allow for use of new or better products.

This design change does not change the Technical Specifications or Bases or other operational requirements. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 12.3-4, Alarm Capability for Area Radiation Monitors (ARMs)

Description

This departure revises FSAR Tables 12.3-3, 12.3-6 and 12.3-7 to add alarm capability to certain area radiation monitors (ARMs). Five additional monitors have been added to the Reactor Building as indicated in Table 12.3-3 and as shown on Figures 12.3-56, 12.3-57, 12.3-58, 12.3-60 and 12.3-62.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1, Tier 2* information, the Technical Specifications, any underlying design or other operational requirements.

This departure represents a favorable change which provides additional alarm capability to area radiation monitors in the reactor building, radwaste building and turbine building, and adds additional area monitors in the reactor building beyond those identified in the DCD. As such,this departure provides additional notification to plant personnel regarding high radiation levels in these areas which is a safety improvement. There is no change in design or function of any other SSC important to safety. By providing additional notification to plant personnel for conditions which could lead to accidents, this departure has no adverse impact on the likelihood or consequences of analyzed accidents or malfunction of an SSC important to safety and may have a favorable impact. Furthermore, it does not change any plant physical features, SSCs important to safety or fission product barriers.

Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not adversely affect any feature for mitigation of an ex-vessel severe accident and may have a favorable effect. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario.

Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 14.2-1, Control Rod Drive Friction Testing Requirement

Description

The DCD Subsection 14.2.12 requirement for performing control rod drive (CRD) friction testing at rated pressure is deleted. CRD friction testing is a traditional requirement performed on older BWR designs with CRDs positioned using hydraulic pressure. The ABWR employs a design in which normal rod positioning is accomplished by an electric motor. Mechanical binding (friction) of an ABWR CRD will result in blade separation from the ball nut which would be detected by permanently installed instrumentation. Thus ABWR CRDs are easily monitored for performance degradation during normal CRD withdrawal and periodic friction testing is not required.

Evaluation Summary

This departure to remove the DCD-required CRD friction testing from the STP 3 & 4 FSAR has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change any Tier 1 or Tier 2* DCD information, the Technical Specifications, Bases for Technical Specifications, any underlying design, or other operational requirements. It does not change any plant physical features, SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 18.4-1, Main Generator Synchronization Control Relocation

Description

The location of the controls and displays required for the synchronization of the main generator is not necessary at the main control console as stated in DCD Tier 2 Section 18.4.2, therefore these have been relocated to the main control room panel. This change allocates space on the main control console to more critical tasks and allows manual synchronization of the main generator by the control room operator or automatic synchronization by the Power Generation Control System. The relocation is consistent with the description provided in DCD Tier 1 Section 2.7.1 and the requirements in Section 18.7 for the final design of the main control room.

Evaluation Summary

The change does not affect any Tier 1 or Tier 2* DCD information. This departure to relocate the controls for synchronization of the main generator to the main control panel has been evaluated pursuant to and determined to comply with the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. This departure does not change the Technical Specifications or Bases of Technical Specifications, any underlying design or other operational requirements. Furthermore, it does not change SSCs important to safety or fission product barriers. Any previously evaluated accident is not affected, and the possibility for an accident of a different type is not created. Also, it does not affect any method used for evaluation in establishing the design bases or in the safety analyses. This departure does not affect any feature for mitigation of an ex-vessel severe accident. For the same reason, and because there is no effect on any event, operation or SSC function, the change does not create a different ex-vessel accident scenario. Therefore, this change has no adverse impact and does not require prior NRC approval.

STD DEP 19.3-1, Evaluation of Common Cause Failures

Description

ABWR Standard Safety Analysis Report (SSAR) Chapter 19.D.8.6 documents the results of a PRA sensitivity analysis on common cause failure of selected mechanical systems performed by GE in response to a request from the NRC during the ABWR certification process. The final paragraph in SSAR Chapter 19.D.8.6 summarizes the results of the sensitivity analysis and indicated that the common cause factors evaluated will be added to the plant PRA model in any future revised basic quantification of the ABWR. The common cause factors were added to the ABWR plant model used to quantify the effects of plant-specific factors for South Texas Project Units 3 & 4. The addition of the common cause terms represents a departure from the PRA described in the reference DCD.

Evaluation Summary

This departure has been evaluated and pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5, as described previously. There is no impact on any Tier 1, Tier 2*, technical specifications, basis for technical specifications or operational requirements as a result of this change. As a result of this departure, there is no effect on the frequency or consequences of any accidents or the likelihood or consequences of malfunctions of SSC important to safety previously evaluated in the DCD. There is no possibility of a new type of accident, and there is no impact on fission product barriers or ex-vessel severe accident events. Therefore, the change has no adverse impact and does not require prior NRC approval.

Based on this evaluation, prior NRC approval of the change is not required.

STD DEP 19.7-1, Control Rod Drive Improvements

Description

Subsection 19.7.2, item 4 of the reference ABWR DCD discusses Control Rod Drive Improvements incorporated into the ABWR design. The second paragraph indicates that the Fine Motion Control Rod Drive (FMCRD) brake design had to be fully testable on an annual basis, presumably during refueling outages because testing of the brakes during power operation is not practical. A clarification is made for consistency with the outages on the 18-month cycle basis for the plant. Words, “an annual,” are replaced with “refueling cycle.” Technical Specification LCO 3.10.12 controls removal of CRD subassemblies during refueling.

Evaluation Summary

This departure has been evaluated pursuant to the requirements in 10 CFR 52, Appendix A, Section VIII.B.5. There is no impact on any Tier 1, Tier 2*, DCD, technical specifications, basis for technical specifications or operational requirements as a result of this change.

The change is required to reflect that the plant refueling outage will be every 18 months, during which the FMCRD brakes can be tested. It does not affect the brake design or function. The FMCRD electro-mechanical brake is a Class 1E safety-related component with a 10-year Environmental Qualification replacement life. Brake performance characteristics testing is performed every 10 years when a replacement/new brake is installed. Thirty-five motor subassembly units, including the brake, will be tested during the 18-month refueling outages. This is sufficient to assure that the brake performance to prevent rod ejection is not affected as considered in the ABWR PRA studies. Section 15.4.9.1 of the DCD characterizes the probability of the initial causes of the control rod ejection accident as low enough to warrant it as a limiting fault. In addition, both the safety-related check valve and brake (see DCD Tier 2, Section 4.6.2.3.3.1.2 and 15.4.9.1) would have to fail in order for a rod ejection to occur. This makes the control rod ejection accident an extremely low probability event. Consequently, there is negligible impact on the probability or consequences of an accident or malfunction of an SSC important to safety.

The FMCRD brake has not been identified as a design feature in the DCD for mitigating an ex-vessel severe accident. The change to the brake testing frequency description does not impact the brake design or function, and therefore, the likelihood or consequences of a severe accident is not impacted.

Based on this evaluation, prior NRC approval of the change is not required.

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

Description

Appendix 19I of the reference ABWR DCD discusses the seismic margins analysis that evaluated the capability of the plant and equipment to withstand a large earthquake of two times the safe shutdown earthquake. Section 19I.7 of the DCD states that since the Atmospheric Control System crosstie valves are normally closed motor-operated valves, that this containment bypass path need not be included in the PRA analysis. This analysis has been changed in the STP 3 & 4 FSAR to reflect the design of air-operators on these valves and, as a result this analysis is the same as for the main purge valves.

Evaluation Summary

This departure has been evaluated pursuant to the requirements of 10 CFR 52, Appendix A, Section VIII.B.5.

There is no impact on any Tier 1, Tier 2*, technical specifications, bases for technical specifications or operational requirements as a result of these changes.

Changing the design input assumption used in the seismic margins PRA analysis as it relates to the design of the ACS crosstie lines / valves is a correction of the PRA analysis basis, and has no effect on the plant design or safety analysis. It has no effect on any plant design or safety analyses. Thus this departure does not have any effect on the frequency of occurrence or consequences of accidents or malfunction of SSC important to safety previously analyzed.

This change to the PRA analysis basis for the ACS has no effect on the design of any systems involved in mitigation of any ex-vessel severe accidents, therefore the likelihood or consequences of an ex-vessel severe accident is not impacted.

As a result of this evaluation, prior NRC approval of the change is not required.

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

Description

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

The ABWR DCD (Chapter 19 and Appendix 19R) was written with the assumption that vertical wet-pit type pumps would be used in the RSW System design. The ABWR DCD addressed the possibility that the UHS basin water could be siphoned into the control building. The return lines contained vacuum breaker valves located on the header that provided water to the cooling towers.

Evaluation Summary

The RSW System is now designed with horizontal split case type pumps to increase system availability and reliability. There is no possibility of siphoning on the supply side with this design, because the RCW heat exchanger supply piping is constantly under positive pressure due to the normal operating hydrostatic head above the pump centerlines. Vacuum breaker valves are ineffective with this piping configuration and will not provide any protective measure against internal flooding of the control building. The vacuum breaker valves on the return piping from the RCW heat exchangers are deleted from the RSW System. The horizontal return piping routed inside the UHS basin is substructured in concrete and the vertical risers are encased in concrete. The vertical risers emerge out of the concrete at an elevation that is above the normal operating water level and are routed to the cooling towers' spray header interface point. Vacuum breaker valves on the return piping do not provide any protective measure against control building flooding due to siphoning.

There are redundant safety-related active motor-operated valves in the supply piping to each of the three RCW heat exchangers. Even if one of these valves is postulated to fail in the open position, there is another motor-operated valve that automatically closes on detection of a high-high water level in the RCW heat exchanger room to prevent gravity drainage from the UHS basin to the control building. There are leak detection measures in the control building that would annunciate and require operators to investigate potential flooding as well as trip the affected division's RSW pumps and close redundant supply side motor-operated valves. Relocation of the UHS and RSW Pump House results in a significant reduction of the stored water volume in the buried RSW piping. This reduction in stored water would result in considerably less water mass that could flow into the control building due to a postulated moderate energy line crack. Consequently, there would be a lower flooding potential to the non-affected RSW divisions due to a lower water level in the RSW division postulated with the moderate energy line crack.

This departure has been evaluated and determined to comply with the requirements in 10CFR52, Appendix A, Section VIII.B.5, as described previously. The change has no adverse impact and does not require prior NRC approval.

STD DEP Vendor, Vendor Replacement

Description

The reference ABWR DCD was developed with numerous statements that activities during construction and startup would be performed in accordance with GE approval or oversight. The intent of these statements was to ensure that the designer was appropriately involved in startup testing or construction activities.

Since the DCD was developed, other vendors have surfaced that have equivalent capability. This standard departure replaces the terms such as GE, GEH, and General Electric with the generic term NSSS Vendor, with an alternative vendor specified, or in some cases has eliminated the term altogether. This departure also replaces General Electric Company's product references such as NEDEs and NEDOs with the corresponding reference of another ABWR vendor whose reference has been approved by the NRC for use in this application. In all cases, the intent of the reference ABWR DCD statement is preserved by the departure and the replacement vendor must be fully qualified to perform the function by STPNOC. Furthermore, this departure only applies to Tier 2 information.

Examples within the scope of this departure include:

During the construction cycle and the various testing phases, additional staff is supplied by the plant owner/operator, the NSSS vendor*, and others.*

For automatic start tests, in order to provide margins to overspeed and isolation trip setting, the transient start first and subsequent turbine speed peaks shall not exceed the requirement specified by the Startup Test *Specification.*

All fabrication of the reactor pressure vessel was performed in accordance with Toshiba - *approved drawings.*

Evaluation Summary

This generic departure does not change the design, the method of controlling the design, or the manner in which the ABWR will be operated. It only changes the use of the words GE, General Electric, GEH etc., to allow the use of other qualified vendors to perform certain functions that are explicitly called out in the reference ABWR DCD. Numerous vendors throughout the world have developed the capability to perform analyses, design and construction in a wide range of nuclear applications. As stated previously, any replacement vendor will be qualified to perform the function by STPNOC.

For all of the above reasons, this change has no impact on the probability or consequences of occurrence of accidents previously evaluated. It does not have an impact on probability and consequences of a malfunction of an SSC important to safety previously evaluated in the DCD. This change in and of itself does not result in a change in a design basis limit for a fission product barrier, nor does it impact a method of evaluation. Likewise, there is no impact on the probability or consequences of a severe accident.

In summary, this standard departure has been evaluated pursuant to the requirements in 10CFR52, Appendix A, Section VIII.B.5. Since this departure applies to Tier 2 information only, there is no impact on any Tier 1 or Tier 2* DCD, Technical Specifications, Basis for Technical Specifications or operational requirements as a result of these changes.

Therefore, this generic change has no adverse impact and does not require prior NRC approval.