2.0 Departures Requiring Prior NRC Approval

This section contains the departures and exemptions from the reference ABWR DCD that require prior NRC approval. These are changes to Tier 1,Tier 1 ITAAC, Tier 2* items (those listed in the tables in "Introduction to the Design Control Document" and marked with italics within brackets), Technical Specifications, and other operational requirements. These changes are numbered according to an assigned primary reference ABWR DCD section.

Section 2.1 contains Tier 1 and Tier 2* departures that are associated with changes from the design as described in the DCD. Some of these changes also require supporting changes to the text of the Technical Specifications.

Section 2.2 contains those changes that affect the Technical Specification that are not caused by the Tier 1 departures in Section 2.1.

These changes are proposed under the rules in 10 CFR 52 Appendix A, section VIII "Processes for Changes and Departures."

The following Tier 1 departures were evaluated against the requirements of 10 CFR 52, Appendix A Section VIII.A.4 and found to be acceptable for exemption. The Tier 2* departure requires prior NRC approval but does not require an exemption.

The following factors apply to each of the Tier 1 departures:

- the design change will not result in a significant decrease in the level of safety otherwise provided by the certified design,
- the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security
- special circumstances are present as specified in 10 CFR 50.12(a)(2); and/or
- the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.

A summary of the above factors applicable to each Tier 1 departure is provided in the written evaluation section for each.

The departures from the generic Technical Specifications have been evaluated against the requirements of 10 CFR 50.12 and found acceptable for exemption. For each of the following departures:

- the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and
- special circumstances are present as specified in 10 CFR 50.12 (a) (2).

STPNOC is not making any departures from the other operational requirements in the reference AWBR DCD and did not identify any departures from Tier 2 that require NRC approval under the criteria in Section VIII.B.5 of Appendix A to 10 CFR 52.

2.1 Tier 1 and Tier 2* Departures from the DCD

The following Tier 1 and Tier 2* departures result from a change in the design described in the DCD.

STP DEP T1 2.1-1, SRV Setpoints and Simmer Margin

Description

The Reactor Safety/Relief Valve (SRV) relief and safety analytical limits and setpoints have been modified relative to Section 2.1.2 of the reference ABWR DCD (Tier 1 change) and also in DCD Sections 5.2 and 15.1 (Tier 2 changes).

These departures have been made for the following reasons:

Tier 1 departures:

- The setpoints and analytical limits have been increased to assure SRV simmer margin is not less than 15%. Simmer margin is the difference between the normal operating static pressure and the SRV direct actuation pressure based on nominal settings. The specific nameplate spring pressure and ASME rated flow rates at 103% of spring set pressure are increased for all valves.

Associated Tier 2 departures:

- The RCPB over-pressure protection analysis is revised to reflect the flow rates for the design SRVs rather than the SRVs described in the reference ABWR DCD.
- The reseating pressure (percent of spring setpoint) is revised to 96 - 90% to reflect the hardware capability. A requirement that the first two valve groups must reset at a value less that 92% is imposed as a result of the transient analysis performed to confirm that the containment hydrodynamic loads assumption of a single valve opening can occur multiple times after the initial multi-valve pop.
- The drift values are revised (increased) for both the safety and relief functions and are adjusted to 3% of the analytic limit to reflect operating experience. Note that all operating BWRs in the US that use standard Technical Specifications have adopted this 3% reset requirement for testing of SRVs.

The changes are required to meet the assumptions in the containment design and to reflect intended hardware design choices. They also maintain consistency with best practices in the US nuclear industry to increase system performance margin and to reduce maintenance requirements. This departure is expected to reduce unnecessary SRV testing during outages.

Evaluation Summary

This evaluation covered Tier 1 and Tier 2 departures.

- The changes to setpoints, reseating pressure, and drift values improve the reliability of the Safety Relief Valves.

- Analysis confirms that both reactor over pressurization and containment loading analyses are not adversely affected by the changes.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement based on BWR operating experience with SRVs. The analytically driven changes are the result of operating history and methodology refinement and therefore will result in a benefit to the public health and safety.

(4) The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. Specifically, the design change represents an improvement in safety, and does not affect the configuration of the plant or the manner in which the plant is operated. Therefore, the reduction in standardization resulting from the change in the setpoints and other values should not adversely affect safety.

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

STD DEP T1 2.2-1 Control Systems Changes to Inputs, Tests, and Hardware

Description

Minor changes are made to the safety-related and non-safety-related control systems as described in the reference ABWR DCD. Design detailing, operational experience, technological improvement efforts, and the desire for clarity prompt the changes to the specific wording in Tier 1 section 2.2.

The reference ABWR DCD provides for an input to the Reactor Protection System (RPS) from Turbine First Stage Pressure. This protection sensor and signal will be replaced with a simulated thermal power signal of reactor power from the Neutron Monitoring System (NMS). The NMS will provide an accurate input of reactor power, regardless of the steam flow or steam bypass paths from the Main Steam System, that are not measured by first stage turbine pressure. This parameter provides a power threshold for a reactor trip on closure of a turbine trip valve or turbine control valve. This change will reduce installation costs, hardware failures, and operating costs over the life of the plant.

This departure provides for reactor power input to the RPS logic that generates a reactor trip signal on Turbine Stop Valve and Turbine Control Valve closure when reactor power is greater than 40%. The input from turbine first stage pressure will be eliminated, and replaced with a signal from the Simulated Thermal Power signal from the Neutron Monitoring System. The replacement of the input from turbine first pressure with a signal from neutron monitoring will be more accurate and more reliable and hence an improvement. With past and current experience of this logic, operating experience shows the benefit of this change to the plant design. This design has operational experience at Leibstadt (Switzerland - BWR 6), which was implemented approximately 20 years ago.

Without the four pressure sensors for turbine first stage pressure, the failure of these components is eliminated as a possible fault input to RPS for the reactor trip logic. Using the Neutron Monitoring System provides more accurate measurement of reactor power, by eliminating the steam flow paths that bypass the turbine (e.g., steam driven pumps, steam jet air ejectors, steam re-heaters, etc.). This method of reactor power measure input to RPS eliminates the mechanical measurement and possibility of mechanical failure of the turbine first stage instrumentation.

The original DCD Tier 1 Section 2.2.1 did not identify that the rod withdrawal block function associated with detection of the separation condition of a control rod is only applicable when the RPS Mode switch is in Startup Mode or Run Mode. Also, DCD Tier 1 Figure 2.2-1 did not identify that the RCIS receives more than the Refuel Mode status signal associated with the RPS Mode switch status (i.e. RCIS actually receives separate status signals associated with each of the four possible RPS Mode switch status conditions: 1) Shutdown Mode, 2) Refuel Mode, 3) Startup Mode and 4) Run Mode). When the RPS Mode switch is in Shutdown Mode, a rod withdrawal block is activated for all control rods; therefore, implementation of an individual rod withdrawal block based upon detection of the separation condition is not necessary.

When the RPS Mode switch is in Refuel Mode, it is only possible to withdraw: 1) one operable control rod or 2) the one or two control rods associated with a single HCU when the Scram Test mode of RCIS is active. All other operable control rods must remain fully inserted (and RCIS interlock logic enforces this situation). Thus, the RCIS logic insures the reactor remains in the subcritical condition regardless of the position of the one or two control rods that can be withdrawn with the RPS mode switch in Refuel Mode. When performing the FMCRD coupling check surveillance test in Refuel mode (for one or two control rods that have been withdrawn), the separation status will be activated when the FMCRD ball nut is withdrawn to the over travel out position. It is required that the separation rod withdrawal block not be activated to allow completion of this required surveillance test. Therefore, implementation of an individual rod withdrawal block based upon detection of the separation condition is also not desired and not necessary when in the Refuel Mode.

Design detailing efforts have located the non-safety-related Feedwater Control System microprocessor-based equipment in both the Turbine Building and the Control Building. Therefore, the DCD Feedwater Control System Tier 1 description is changed to delete the sentence stating, “The FDWC System microprocessors are located in the Control Building.”

The reference ABWR DCD Tier 1 Table 2.2.1 ITAAC Acceptance Criteria for Item 11 (i.e. associated with testing of one of the dual redundant non-Class 1E uninterruptible power supply at a time) states the “test signal exists in only one channel at a time.” This acceptance criterion was based upon an assumption that in the RCIS design implementation each channel of the dual-redundant RCIS controller equipment would receive power from only one associated uninterruptible power supply. However, in the final RCIS design implementation, only the power supply associated with the one non-Class 1E uninterruptible power supply being tested will become inoperable and both of the dual-redundant controller channels remain operational when this testing is conducted. The detailed RCIS design for the dual-redundant controller equipment is implemented such that each channel remains operational as long as either one of the uninterruptible power supplies is operational. There is an associated alarm condition activated when one of the uninterruptible power supplies becomes inoperable (i.e. so the operator becomes aware of this abnormal power supply status condition). A change has been incorporated regarding the DCD Tier 1 ITAAC requirement for the RCIS related to the Acceptance Criteria associated with the testing of one of the dual redundant non-Class 1E uninterruptible power supply at a time.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and therefore will result in a benefit to the public health and safety.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization, Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

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

STD DEP T1 2.3-1 Deletion of MSIV Closure and Scram on High Radiation

Description

The Scram and MSIV automatic closure on high MSLRM (main steam line radiation monitor) trip is deleted. Elimination of these functions reduces the potential for unnecessary reactor shutdown caused by spurious actuation of the MSLRM trip and increases plant operational flexibility. As a result, this change increases reliability and plant availability and therefore results in a benefit to public health and safety.

This departure includes the following Tier 1, Tier 2 and Technical Specification changes.

Tier 1 departures:

Changes have been made relative to the reference ABWR Tier 1 DCD Figure 2.3.1, "Process Radiation Monitoring System Control Interface Diagram" to remove the MSL Tunnel Area Radiation input from the plant sensors that provide input data.

Tier 2 departures:

Changes have been made relative to the reference ABWR Tier 2 DCD Sections 5.2, 6.2, 7.2, 7.3 and their associated tables to remove information pertaining to main steam line high radiation monitoring and process radiation monitoring system. Section 11.5 has been modified to move main steam line tunnel area radiation monitoring information from the section describing "monitoring required for safety and protection" to the section describing "monitoring required for plant operation." Main steam line high radiation information has also been removed from Tables 18F-1 through 3 and Tables 18H-2 and 18H-5.

Generic Technical Specification departures:

Generic Technical Specifications LCO 3.3.1.1 and LCO 3.3.6.1 and their associated Bases has been modified to remove the Main Steam Tunnel Radiation High function.

The original purpose of this instrumentation was to close the MSIVs in order to mitigate the potential release of fission products released from fuel rods. The initial release of primarily noble gas fission products from the damaged fuel rods was expected to cause a spike in the radiation readings on the steam line which would initiate safety related actions. However, radiation sources in the steam lines are primarily dominated by N-16 emissions, and setpoints sufficient to sense noble gas spikes can be overwhelmed by minor variations in N-16 flow causing spurious trips. Since sensors on the condenser steam jet air ejector and ventilation stack can also serve the purpose of monitoring potential offsite releases, the BWROG LTR NEDO-31400A concluded that the vessel isolation (MSIVs) and scram functions of the MSLRM are not required.

The MSLRM alarms in the main control room and the conclusion of NEDO-31400A section 2.2 remain valid for the STP proposed design. Both the MSLRM and the Condenser Steam Jet Air Ejector (SJAE) monitors will alert the operators for events that can cause an elevated release of radioactivity from the fuel. In addition to these alarms, given sufficient time, the offgas treatment system radiation monitor and eventually the stack effluent monitor will activate alarms in the control room and the operator will be able to isolate the offgas system to stop these releases to the environment. Therefore, even without the automatic reactor shutdown function and the MSIV closure on the MSLRM trip, the operator will be able to limit offsite releases.

For operating plants, the NRC has approved this change (see NEDO-31400A). The following additional considerations apply to the ABWR:

In the NEDO Section 9 there was a discussion of the negligible increase in reactivity control failure frequency with the deletion of the MSLRM scram function. This increase was due to operating plants taking credit for the MSLRM initiating a scram during the control rod drop accident. As described in Tier 2 DCD Section 15.4.10, it is concluded that for the ABWR, there is no basis for the control rod drop event to occur or to perform a related radiological analysis. The deletion of the automatic scram and MSL isolation result in no change in associated risk.

The NRC's approval letter accepting the NEDO requires an applicant referencing it to:

- Demonstrate that the plant design features affecting a rod drop analysis bound those used in the report. For ABWR, since the event has no basis. A comparison of design features affecting the rod drop analysis against those assumed in the report is unnecessary.
- Demonstrate a basis (e.g., plant procedures) to conclude that manual operator action will be taken expeditiously in case of increased levels of radioactivity in the MSLs. STP 3 & 4 alarm response procedures will direct operator action in case of increased radioactivity levels.

Since this change incorporates a BWR change that was previously approved by the NRC and since the SER conditions are met for the ABWR as explained above, there are no adverse effects on plant performance.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and therefore will result in a benefit to the public health and safety.

(4) This is "standard" departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization, Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

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

References

(1) Licensing Topical Report NEDO-31400A "Safety Evaluation for Eliminating the BWR Main Steam Isolation Valve Closure Function and the Scram Function of the Main Steam Line Radiation Monitor," October 1992.

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

Description

The reference ABWR DCD has two RHR loops connected to the Fuel Pool Cooling system with normally closed crosstie valves. During refueling outages, a crosstie valve can be opened to allow direct cooling of the fuel pool by circulation of fuel pool water through the RHR heat exchanger and returning it to the fuel pool. In addition, the RHR pumps have the capability to provide fuel pool emergency makeup water by transferring suppression pool water to the fuel pool. This change is to add the capability to allow the choice of a third loop, RHR division A, in the Augmented Fuel Pool Cooling and Fuel Pool Makeup Modes.

This addition of piping and valves will be of the same quality standard, seismic category, and ASME code as the B and C RHR loops components, along with another capability to provide makeup or cooling to the Spent Fuel Pool. Only one RHR cooling loop will be aligned for the Augmented Fuel Pool Cooling or Fuel Pool Makeup Mode at any one time. The additional loop will increase the reliability from a single failure standpoint. This design change was chosen based on improved reliability and performance.

This change provides the ability to supply fuel pool cooling or makeup from any of the three RHR loops in the Augmented Fuel Pool Cooling or Fuel Pool Makeup Modes. This will enhance capabilities and reliability to perform division outages for maintenance and other activities. Division outages will be better able to be coordinated during all plant operational Modes. During design detailing it was recognized that the added flexibility of having the capability to perform divisional outages in any order was a worthwhile design improvement. As an example, if Division B EDG constitutes a critical path for an outage, in order to maintain a single failure margin, work could not start until core decay heat decreased to the point that RHR Spent Fuel Pooling augmented cooling was no longer required. By having all three divisions capable of supporting the Augmented Fuel Pool Cooling Mode, Divisional Outages (potential critical path) could occur based on workload in the division.

Evaluation Summary

During design detailing it was recognized that the added flexibility of having the capability to perform divisional outages in any order was a worthwhile design improvement. As an example, if Division B EDG constitutes a critical path for an outage, in order to maintain a single failure margin, work could not start until core decay heat decreased to the point that RHR Spent Fuel Pooling Assist was no longer required. By having all three divisions capable of supporting Spent Fuel Pool Cooling assist, Divisional Outages (potential critical path) could occur based on workload in the division.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an increase in redundancy and therefore will result in a benefit to the public health and safety.

(4) This is "standard" departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization, Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

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

STD DEP T1 2.4-2 Feedwater Line Break Mitigation

Description

This departure reduces challenges to the containment pressure design value following a feedwater line break (FWLB). The corrective design concept is a trip of the condensate pumps following an indication that a Feedwater Line Break (FWLB) in the drywell has occurred.

The FWLB is the limiting design basis accident for ABWR primary containment vessel (PCV) peak pressure response. This is because blowdown flows from both the reactor pressure vessel (RPV) side and the balance of plant (BOP) feedwater side contribute to the peak pressure response. Previous BWR designs were bounded by the recirculation line break, that is not a consideration in the ABWR design.

Calculations performed during STP 3 & 4 initial planning as a check against the FSER estimate that the ABWR containment design pressure would be exceeded at the 3926 MWt power level with an injection of approximately 100% feedwater flow at 15 minutes. The licensing basis for ABWR is no operator actions for 30 minutes for design basis accidents. With the current ABWR design, the only mitigation would be operator action using the non-safety trip of the condensate and/or feedwater pumps.

Therefore, high drywell pressure signals already existing in the Leak Detection & Isolation (LDI) logic of the Safety System Logic & Control (SSLC) are used, in conjunction with differential pressure signals between the two feedwater lines, to identify a FWLB in containment and to then trip the condensate pumps.

The departure implementation of condensate pump trip improves plant safety by limiting the mass flow to the drywell after the FWLB, thereby ensuring the predicted peak pressure will not exceed the design value. The instrumentation logic to initiate the trip will be an “AND” circuit to reduce the probability of false trips. That is, the logic will require excessive differential pressure between the two-feedwater lines “AND” high drywell pressure to initiate the condensate pump trip. This will reduce the negative impact on plant operation, plant reliability and availability. There would not be an impact on the PRA by adding circuit breakers for the condensate pump supplies because the logic will only be initiated during FWLB LOCA, the breakers will be normally closed, and additional operator actions will not be required to start the condensate pumps during other events.

Evaluation Summary

These changes ensure that the containment pressure margins are maintained during the limiting containment pressurization accident. Consequentially, the changes decrease the risk associated with the feedwater line break inside containment. These changes maintain the same level of plant reliability and performance as described in the DCD. The changes will provide a better level of plant protection and personal safety and a net benefit to the public health and safety. While this involves changes to an SSC, there are no adverse effects on any DCD design function. No procedure was changed.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and better conformance to licensing criteria (no operator action until 30 minutes) and therefore will result in a benefit to the public health and safety.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization. Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

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

STD DEP T1 2.4-3 RCIC Turbine/Pump

Description

The original DCD incorporated a steam turbine driven water pump that has been historically used in the United States with BWR plants. During the design detailing stage of the ABWR development, another design was chosen based on improved reliability, performance, and simplicity. The new design meets or exceeds all safety-related system performance criteria including start time, flow rate, and low steam pressure operation.

The improved design and system simplification is due to (a) monobloc design (pump & turbine within same casing); (b) no shaft seal required; (c) no barometric condenser required; (d) no oil lubrication or oil cooling system required because the system is totally water lubricated; (e) no steam bypass line required for startup; (f) simpler auxiliary subsystems; and (g) no vacuum pump and associated penetration piping or isolation valves required. The monobloc design is of horizontal, two-stage centrifugal water pump driven by a steam turbine contained in a turbine casing integral with the pump casing. The turbine wheel has a single row of blades. The pump impellers, turbine wheel and inducer are mounted on a common shaft, which is supported on two water lubricated journal bearings. The bearings are housed in a central water chamber between the turbine and pump sections and are lubricated by a supply of water taken from the discharge of the first stage impeller and led to the bearings through a water strainer.

The pump is supported on the pedestals of a fabricated steel base plate by feet formed on the pump casing and central water chamber. The monoblock construction of the pump eliminates the need for alignment between the pump and the turbine. The operating speed of the pump is governed by the turbine control subsystem which regulates the quantity of steam to the turbine based on discharge pressure. The main elements of the control gear are the steam stop valve, the throttle valve and the pressure governor. The pump is also provided with electrical and mechanical over speed trip mechanisms which close the steam stop valve when the speed exceeds predetermined levels. Speed measurement is provided by an electronic tachometer.

One less containment penetration is required and approximately 10 meters of small bore piping previously analyzed for interfacing system LOCAs and upgraded to burst pressure have been removed from the design. The fire loading in the RCIC pump room is reduced by the elimination of the lube oil subsystem and 106 liters of Class III B lube oil.

Licensing Topical Report NEDE-33299P was submitted to the NRC by General Electric Company December 2006 proposing this change as a generic revision to the Design Control Document. More detail on this change may be found in this LTR.

Evaluation Summary

The events in FSAR Chapter 15 were evaluated based on the quickened response times expected so the dynamics of upset and accident responses are not compromised. The events and accidents in Chapter 15 were reviewed. The analyses and conclusions presented in Chapter 15 are not affected. No negative impacts on severe accident probability or severity have been identified nor has a new type of severe accident been created. The bases in the generic Technical Specifications in Chapter 16 will be met or exceeded. This departure results in no negative impact on safety, plant operation or cost. Plant availability and reliability will improve due reduction of active and passive components. Improved turbine reliability well improves plant safety as will improve transient and startup characteristics.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and therefore will result in a benefit to the public health and safety.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization. Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

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

References

(1) NEDE-33299P, "Licensing Topical Report - Advanced Boiling Water Reactor (ABWR) with Alternate RCIC Turbine-Pump Design," December, 2006

STD DEP T1 2.12-1 Electrical Breaker/Fuse Coordination and Low Voltage Testing

Description

The reference ABWR DCD in Tier 1 states electrical power distribution interrupting devices (circuit breakers and fuses) are coordinated such that the interrupting device closest to the fault opens first. The description of the interruption device coordination has been modified to include the acceptable industry practice with standards and codes (e.g., IEEE 141, IEEE 242, etc.). Including this provides detailed guidance for electrical system design expectations. Since protective device coordination may overlap, and the discrete coordination may not be possible, the expectation has been changed to meet the requirement to the maximum extent possible.

The reference ABWR DCD ITAAC also requires that pre-operational/start-up testing of the as-built Class 1E Electrical Power Distribution System will be conducted by operating connected Class 1E loads at their analyzed minimum voltage. DCD Table 2.12.1 (Electric Power Distribution System ITAAC) currently states that tests of the as-built Class 1E Electric Power Distribution System will be conducted by operating connected Class 1E loads at their analyzed minimum voltage. Testing in this manner for each connected Class 1E load is not practical to connect and disconnect each load, one at time to facilitate testing.

For DC loads, ITAAC require testing by operating connected Class 1E loads at both the minimum and maximum battery voltages. Tier 1 DCD Table 2.12.12 (Direct Current Power Supply ITAAC) currently states that tests of the as-built Class 1E DC system will be conducted by operating connected Class 1E loads at less than or equal to the minimum allowable battery voltage and at greater than or equal to the maximum battery charging voltage. It is not practical to perform testing in this manner. This is modified to allow performance type tests at the manufacturer's shop for the operating voltage range of Class 1E AC and DC electrical equipment prior to shipment to the site. In addition, system preoperational tests will be conducted on the as-built Class 1E AC and DC systems and test voltage results will be compare against system voltage analysis.

Evaluation Summary

For electrical loads powered at or below 120 VAC or 125 VDC, the requirement that the device closest to the fault open first is not always met, since many small loads have internal fuses/circuit breakers and there is often a minimum device size available, or the minimum circuit breaker/fuse size recommended by the vendor. In the case of high fault current, the upstream protective device may trip before the protective device connected to the small load, or both may trip at the same time. In such cases, discrete coordination may not be possible.

The extensive in-situ testing in the DCD is not necessary and is duplicated, since the voltage tests are performed by the manufacturers as part of their normal performance and functional tests prior to shipment. In addition, testing is performed at the jobsite on electrical power distribution equipment during construction after its installation.

The events and accidents in Chapter 15 were reviewed. The analyses and conclusions presented in Chapter 15 are not affected as the alternate methods of breaker coordination and low voltage testing are judged equivalent to those in the DCD. No negative impacts on severe accident probability or severity have been identified nor has a new type of severe accident been created. The bases in the generic Technical Specifications in Chapter 16 will be met or exceeded.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the change is intended to accomplish the same purpose as the original DCD design and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since the change accomplishes the same underlying purpose as the original DCD design.

(4) This change is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization.

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

References

(1) IEEE 141-1993, Recommended Practice for Electric Power Distribution for Industrial Plants (IEEE Red Book)

(2) IEEE 242 -2001, Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems (IEEE Buff Book)

STD DEP T1 2.12-2 I&C Power Divisions

Description

A fourth division of safety related power has been added to the Class 1E Instrument and Control Power Supply System.

The Instrument and Control Power Supply System as described in the DCD Tier 1 provided power to three mechanical safety-related divisions (I, II and III) and not to safety-related Distributed Control and Information System (DCIS) Division IV. This departure adds a fourth regulating transformer and associated distribution panels to supply Instrument and Control Power to Division IV.

The DCIS cabinets and chassis, ECCS Digital Control and Information System cabinets and chassis, in each of the four divisions, use redundant power supplies and feeds for increased reliability and availability to allow self-diagnostics and to operate during power failures. The existing design provides three divisions such that the two feeds are uninterruptible vital AC power (uninterruptible does not mean single failure proof) and I&C power (interruptible but diesel-backed). The second I&C power feed is available to the Division IV DCIS cabinets and chassis. Most power problems can be addressed on-line and all such problems will be "non-critical" faults since no functionality will be lost.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and therefore will result in a benefit to the public health and safety.

(4) This is "standard" departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization. Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

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

STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

Description

The departure relative to the reference ABWR Tier 1 and Tier 2 DCD and Generic Technical Specifications is documented in detail in Licensing Topical Report (LTR) NEDE-33330P, "Hydrogen Recombiner Requirements Elimination", March 2007, proposing this change as a generic revision to the Design Control Document. More detail on this change may be found in the LTR.

The reference ABWR DCD requires two redundant hydrogen recombiners and safety-related hydrogen/oxygen analyzers. This includes associated containment isolation valves, safety-related cooling water, and Class 1E power supply. This departure removes the hydrogen recombiners and associated components. The hydrogen and oxygen monitors are retained but downgraded from safety-related to non-safety-related. This change will not affect the Containment Spray System and the mixing it provides to prevent oxygen pockets.

Amended since DCD issuance, 10 CFR 50.44, "Standards for Combustible Gas Control System in Light Water-Cooled Power Reactors", does not currently require light water reactors, operating with an inerted containment, to have hydrogen recombiners. With this rule change, the recombiners and hydrogen monitoring equipment no longer meets any of the criteria in 10 CFR 50.36(c)(2)(ii) for retention in the Technical Specifications and are removed from Chapter 16 and Part 4.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change conforms to current regulations and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii), and special circumstance (vi) are invoked as evidenced by the revision to 10 CFR 50.44 as the underlying purpose is still served and the revision of regulations is a material change of circumstances.

(4) This is "standard" departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization.

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

References

(1) Licensing Topical Report NEDE-33330P, "Hydrogen Recombiner Requirements Elimination" March 2007

(2) 10 CFR 50.44, "Standards for Combustible gas Control System in Light Water-Cooled Power Reactors"

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

Description

This departure modifies the design of certain devices, functions and standards related to the Essential Multiplexing System (EMS) and Safety System Logic and Controls (SSLC). In general, this collection of changes enable the descriptions of the EMS and SSLC to be modified in such a way that they describe integrated top-level functions with a set of supporting sub-functions, as opposed to the current DCD descriptions, which describe them as separate systems with single-purpose hardware components.

The reference ABWR DCD design descriptions reflect outdated technology and are inconsistent with currently available systems and equipment. This change deletes references to components such as Control Multiplexing Units (CMUs), Remote Multiplexing Units (RMUs) and others that refer to an outdated technology (multiplexing), and imply hardware components (units) with limited purposes. In the text changes in FSAR chapter 7 and elsewhere, equivalent data communication functions are described, but not in the context of "multiplexing" or specific hardware components.

This departure also enables specific architecture changes in the Engineered Safety Functions (ESF) portion of the I&C architecture. Specifically, it limits the application of dual redundant Safety System Logic Units (SLUs) and 2-out-of-2 output voting only to those situations where the physical system arrangement and consequences of inadvertent actuation of equipment warrant such protection against inadvertent actuations. Also it eliminates SLU channel bypass function in this same portion of the structure. Also it allows the use of multiple processors where figures imply single processors.

This departure also deletes or supplements references to specific outdated communication protocol standards (both Tier 2* and Tier 2.) In the ten plus years that have passed since the reference ABWR DCD was finalized, network technologies have evolved to the point where the concepts and hardware described in the DCD are no longer available in modern, commercially available networks.

At the time the GE ABWR was certified, the detailed design of the I&C equipment was not established. GE recognized that with the rapid evolution of the I&C technology, the preferred I&C design, including the design of equipment implementing the logic of ESF systems, would almost certainly include modules, components and capability not yet envisioned. Consequently, a specific equipment design for the SSLC systems was not established.

Even though no specific I&C equipment design was established for the ABWR, GE understood that the ABWR would make more extensive use of digital equipment than any previous plant design. GE developed a design of the SSLC and the supporting Essential Multiplexing System (EMS) using then current technology. These were structured as separate systems.

This generalized design, particularly for the EMS, was necessary to establish fundamental architectural elements, provides for the adequacy of diversity, and establishes a sufficiently comprehensive set of standards to be applied for the actual detail design. Specifically, such aspects as the system architecture constraints, system and software design processes, and equipment qualification requirements were established. These specific requirements relative to the I&C divisional architecture, inclusion of hardwired backups for diversity, specific design process related standards to be followed, and specific EMS redundancy requirements became the ABWR DCD. All of those basic requirements are unaffected by the changes covered by this report, but in some cases the requirements are met with a different SSLC architecture.

GE established the architecture for the ABWR for both the RPS and ESF portions of the SSLC based on NUMAC-type equipment. NUMAC-based I&C typically uses one NUMAC chassis per division for the system unless the total input/output count is too large, or the total computational load exceeds the capacity of the NUMAC chassis processing modules, or there is a specific need to maintain operability of part of the channel with equipment out of service. At that time, multiplexed processing of data was typically handled with somewhat independent "multiplexing systems". Based on implementation with NUMAC-type equipment, the potential loading for specific processors, and a separate multiplexing system, the design divided the SSLC into such sub-parts as digital trip modules (DTMs), trip logic units (TLUs), safety system logic units (SLUs) and remote multiplexing units (RMUs) and included a separate EMS. The design for ESF included pairs of SLUs in each division. In addition to the SSLC design, a design of the EMS was established based on the then-available methods and standards in the rapidly advancing area of multiplexing. The ABWR certified design was used to evaluate overall system issues and was the basis for PRA evaluations.

The DCD requirements included: 1) I&C divisions and system logic assignment; 2) divisions of sensors (typically all four divisions); 3) divisions of actuators (three divisions for ESF systems); and 4) EMS architecture having redundancy within each division. The ABWR supporting Tier 2 material also included the full safety design bases for all of the SSLC-related systems and a description of how the General Design Criteria (GDC) are satisfied. None of these requirements is affected by the changes covered by this departure.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents another method for accomplishing the same purpose and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since the design change represents an another method of accomplishing the underlying purpose of the DCD.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization.

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

STP DEP T1 5.0-1 Site Parameters

Description

The site parameters in the reference ABWR DCD were selected to bound most potential US sites. However, the STP 3 & 4 site, when site historical data is analyzed to using current methodologies and standards, represents three specific increases from the generic envelope.

The site design basis flood level is increased from that specified in the DCD. The certified design site parameter for site flooding is changed from 30.5 cm below grade to 414.5 cm above grade (1036.3 cm above mean sea level (MSL)) in order to handle a main cooling reservoir failure as a design basis event at STP.

The main cooling reservoir at the South Texas site is a non-seismic category 1 dam; hence, its failure must be assumed in the worst possible location. This results in the site design basis flood. The maximum flood level is 1450.8 cm above MSL; however it decreases with distance from the main cooling reservoir.

STP 3 & 4 safety-related SSCs are designed for or protected from this flooding event by watertight doors to prevent the entry of water into the Reactor Buildings and Control Buildings in case of a flood. Exterior doors located below the maximum flood elevation on the 12300 floor of the Reactor Building and Control Building are revised to be water-tight doors. The Ultimate Heat Sink storage basin and the RSW pump houses are water-tight below the flood level.

The maximum design precipitation rate for rainfall at the STP site is calculated to increase from 49.3 cm/hr to 50.3 cm/hr based on site meteorology studies. This value is one factor in determining the structural loading conditions for roof design. ABWR Seismic Category 1 structures have roofs without parapets or parapets with scuppers to supplement roof drains so that large inventories of precipitation cannot accumulate. Therefore, the increase in maximum rainfall rate does not result in a substantial increase in the roof design loading, and therefore does not affect the design of these structures.

The humidity at the STP 3 & 4 site, as represented by wet bulb temperature, is increased from that specified in the DCD.

		DCD	STP 3 & 4
Wet Bulb 1% Exceedance Values			
	Maximum Coincident	25°C	26.3°C
	Maximum Non-coincident	26.7°C	27.3°C
Wet Bulb 0% Exceedance Values (historical limit)			
	Maximum Non-coincident	27.2°C	29.1°C

The maximum dry-bulb temperature in combination with coincident wet-bulb temperature provides the state point (enthalpy of the air) that is used as design input for HVAC system design to determine cooling loads. The 1% exceedance STP site-specific state point value is not bounded by the 1% exceedance ABWR state point value.

The Control Building HVAC, Reactor Building Secondary Containment HVAC, and Reactor Building Safety Related Electrical Equipment HVAC systems are designed for an outdoor summer maximum temperature of 46°C. This temperature corresponds to the ABWR 0% exceedance value. The ABWR 0% exceedance state point bounds the STP site-specific 0% exceedance state point and the 1% exceedance state point. The reference ABWR DCD cooling loads calculated based on 0% exceedance values for Control Building HVAC, Reactor Building Secondary Containment HVAC, and Reactor Building Safety Related Electrical Equipment HVAC systems are bounding. Therefore, the change in 1% exceedance coincident wet bulb temperature has no adverse impact on these HVAC systems.

The Radwaste Building HVAC systems have been redesigned using STP site-specific ambient temperatures and the revised HVAC design is compliant with STP 3 & 4 Characteristics.

The maximum non-coincident wet-bulb temperature is used as input for short-term performance of cooling towers and evaporative coolers. In the case of STP 3 & 4, this value is an hourly data point. The site-specific maximum non-coincident wet-bulb temperatures on an hourly basis are not bounded by the reference ABWR site parameters. However, the calculated 30-day and 24-hour consecutive maximum non-coincident wet-bulb temperatures have been determined to be less than the reference ABWR DCD non-coincident hourly value. The UHS cooling tower long-term cumulative evaporation for the postulated LOCA case has been evaluated using the STP site-specific worst-case 30 consecutive day weather data. The UHS basin water temperature has been evaluated using the worst one-day (24 hour) weather data. Thus, the 0% exceedance and 1% exceedance values for non coincident wet-bulb temperatures not being bounded have no adverse impact on the STP 3 & 4 UHS analysis.

Evaluation Summary

These changes establish an equivalent level of site reliability and performance as described in the DCD.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change will maintain the level of safety otherwise provided by the design.

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

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, the remedial measure of water-tight doors provides a net increase in public safety relative to the design specified in the DCD, satisfying special circumstance (iv). Additionally, the changes qualify for special circumstance (ii) in that the changes are intended to accomplish the underlying purpose of the DCD, namely to ensure that the design is able to withstand natural phenomena. Further, special circumstance (vi) is present in that material circumstances not considered during the ABWR certification was granted in location and meteorological history analysis techniques. Given the need for power in Texas, it is in the public interest to allow construction of additional reactors at the STP site.

(4) The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. Specifically, the design change of adding water-tight exterior doors represents an improvement in safety, and does not affect the configuration of the plant or the manner in which the plant is operated. Therefore, the reduction in standardization resulting from the change should not adversely affect safety.

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

STD DEP 1.8-1, Tier 2* Codes, Standards, and Regulatory Guide Edition Changes

Description

Tier 2, Table 1.8-20 lists reference ABWR DCD compliance with NRC regulatory guides. Table 1.8-21 lists applicability of industry codes and standards. This departure identifies Tier 2* items on these two tables that are being updated to more current revisions/editions. Those Tier 2 items that are explicitly revised in the COLA or require change due to changes in the Tier 2* items are also included.

Regulatory Guide 1.75, “Physical Independence of Electric Systems”, revision 3, dated 2/05; and Regulatory Guide 1.153 “Criteria for Power, Instrumentation, and Control Portions of Safety Systems”, revision 1, dated 6/96 are adopted to ensure b more recent industry design and construction practices are used.

The 1992 edition of IEEE 384 “Criteria for Independence of Class 1E Equipment and Circuits” is adopted. IEEE 603 “Standard Criteria for Safety Systems for Nuclear Generating Stations” is updated to the 1991 version. These editions of the standards are currently endorsed by the NRC.

Mil-Specs for electromagnetic inference analysis and control are updated to more current versions as this field has advanced considerably since certification.

Current approved ASME code cases per Regulatory Guide 1.84, “Design and Fabrication Code Case”, revision 33, dated 8/05 may be used in the future. With this update, Regulatory Guide 1.85, “Materials Code Case Acceptability, ASME Section III, Division 1" on ASME material code cases is obsolete and has been deleted as it is now incorporated into revision 33 of R.G. 1.84.

The American Concrete Institute code ACI 349 is updated to the 1997 edition. The ASME Section III Division 2 is updated to the 2001 edition with 2003 Addenda. These combined recognize advances in earthquake engineering and allows efficient use of modularization during construction. Note that ASME Section III Division 1 for piping is not changed from the 1989 edition. This departure also updates Tier 2 to refer to Regulatory Guides 1.136, “Materials, Construction, and Testing of Concrete Containments” revision 3 dated 3/07, and Regulatory Guide1.142, “Safety-Related Concrete Structures for Nuclear Power Plants” to revision 2 dated 11/01.

Evaluation Summary

As a Tier 2* departure, this departure requires prior NRC approval. These updates to more current revisions/editions will increase plant reliability and performance by capturing selected advancements in engineering theory and practice since issuance of the design certification. The revisions to the Regulatory Guides are the current ones in force. The revisions to the industrial codes and standards have been approved or endorsed by the NRC. These enhancements will provide the same level of plant protection and personal safety and are a net benefit to the public health and safety. Changes to Tier 2 items are incidental to the Tier 2* changes.

2.2 Departures from the Generic Technical Specifications

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

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

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

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

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

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

STD DEP 4.4-1, Stability Analysis

This departure is for the analysis to support the implementation of the stability Option III in ABWR. This analysis is provided in licensing topical report NEDO-33336, "Advanced Boiling Water Reactor (ABWR) Stability Evaluation," June 2007.

As a result of the LTR, Technical Specification 3.3.1.1 (Actions, Table 3.3.1.1-1 Footnote (c)) and the associated Bases (Applicable Safety Analysis, LCO, and Applicability for Function 2.f, Actions, References) are changed.

Technical Specification 3.3.1.1 Action J.2 has been added as an alternate Action to Action J.1 which provides initiation of an alternate method to detect and suppress thermal hydraulic instability oscillations, and Table 3.3.1.1-1 Footnote (c)) provides the periodicity for the period-based algorithm (PBA).

The Bases are changed to show three algorithms, not just two, and that the PBA is the only algorithm credited in the applicable safety analysis, and provides supporting information on the OPRM Function and operability requirements. The Bases are changed to show the addition of Action J.2 and provide a discussion for this Action. The Bases are changed to add reference to BWROG-94079, BWR Owner's Group Guidelines for Stability Interim Corrective Actions, June 1994.

Evaluation Summary

Special circumstance (iv) applies in that the new methodology complies with NRC requirements and so is an improvement to public health and safety. These improved and expanded software algorithms provide more comprehensive detection and dampening of core power oscillations.

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

STD DEP 6.2-1, Containment Purge Valve Resizing

This departure changes the size of the butterfly isolation valves and influent and effluent lines of the Atmospheric Control System to the containment from 550A to 500mm. The changes are necessary and desirable for the STP 3 & 4 design because the valve design and manufacturing technology has improved since the inception of the reference ABWR DCD. This improved and advanced technology provides compact, efficient and fast-closing butterfly valves most suitable for the containment isolation function. The interfacing pipe size and penetration also match the valves.

Technical Specification 3.6.1.3 (SR 3.6.1.3.2, SR 3.6.1.3.14) and its associated Bases (Background, LCO, SR 3.6.1.3.14) are changed to show the valve size change to 500mm.

Evaluation Summary

Special circumstance (iv) applies to this departure because the new valve design is faster acting and represents less area for leakage in case of a failure to close.

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

STD DEP 6.2-2, Containment Analysis

The following changes are included in this departure:

- Reference ABWR DCD Section 6.2 provides results of the containment analysis performed for the certified design. This analysis is being updated in the STP 3 & 4 FSAR to reflect changes in regulatory and industry guidance, provide additional design detailing, and provide increased analytical accuracy.

- The design assumptions for Feedwater Line Break (FWLB) have been updated:

  – The feedwater system side of the FWLB is modeled by adding a time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy are determined from the predicted characteristics of typical feedwater system performance.

  – A mitigation feature is added to the ABWR standard design to ensure the conservatism of the mass flow from the FWLB. A break of the feedwater line is detected by instrumentation that measures the differential pressure between the two feedwater lines and then confirmation of high drywell pressure will enable the logic to trip the condensate pumps thus limiting the mass from the break to that in the condensate discharge and feedwater system piping. The logic and breakers will be safety related to ensure that only safety related equipment is credited in the analysis. The design details are described in Departure T1 2.4-2.

- ANSI/ANS 5.1 1979 sets forth methods for calculating decay heat power from fission products, U^{239} and Np^{239} following shutdown of light water reactors. Calculations performed by GE concluded that additional terms not explicitly included in the ANSI/ANS standard, while individually negligible, when summed together may be non negligible. These summation calculations determined that the inclusion of the additional actinides other than U^{239} and Np^{239} and activation products does not significantly affect short-term decay heat calculations. However, for time after shutdown greater than 10^4 seconds (~ 3 hours), the decay heat can be larger. The decay heat input is revised for the STP 3 & 4 FSAR containment analysis. The revised decay heat analysis is based on GE 14 Fuel which is conservative relative to the licensing basis GE 7 fuel.

The ABWR Containment analysis has been updated to reflect the performance of the horizontal vents configuration that had not been modeled in the DCD.

Technical Specification 3.6.1.1, 3.6.1.2, 3.6.1.4, and 3.6.1.5 Bases (Applicable Safety Analyses) are changed based upon the containment analysis. These changes show the peak containment pressure (P^a) from the containment analysis.

Technical Specification 3.6.1.6 (SR 3.6.1.6.3) and its associated Bases (Applicable Safety Analysis, LCO, Action B.1, SR 3.6.1.6.1, SR 3.6.1.6.3) are changed based upon the containment analysis. These changes show the wetwell-to-drywell vacuum breaker open limit, and the steam condensation cases from the containment analysis.

Technical Specification 3.6.2.4 (SR 3.6.2.4.2) and its associated Bases (Applicable Safety Analysis, SR 3.6.2.4.2) are changed based upon the containment analysis. These changes remove the maximum RHR flow rate from the SR, and reference containment spray versus wetwell spray.

Evaluation Summary

Special circumstance (iv) applies in that this represents a benefit in public health and safety. The more advanced and complete analysis methods and new mitigation features provide a more accurate prediction of peak containment conditions post-accident.

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

STD DEP 7.2-3, MSIV/RPS Interface

Subsection 7.2.1.1.4.2 of the reference ABWR DCD describes implementation of Main Steam Line Isolation Valve status as utilizing a data communication function. STP 3 & 4 FSAR wording provides clarification by deleting the data communication function from the description. This departure ensures a clear description is provided and does not affect any safety function. A hardwired signal path is provided to meet timing requirements that cannot be met with the distributed digital system.

Technical Specification 3.3.1.2 Bases (Applicable Safety Analysis, LCO, and Applicability) is changed to show the nomenclature change in the instrument and control architecture that changes Digital Trips Module (DTM) to Digital Trip Unit (DTU).

Evaluation Summary

Special circumstance (iv) applies in that the new hard-wired link is faster acting and more reliable than a network-based communications line. It is also less dependent on support systems like instrument power. Hence it represents an improvement in public health and safety.

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

STD DEP 7.3-3, RPV Level Measurement

Subsection 7.3.1.1.1.2(3)(a) of the reference ABWR DCD describes the Automatic Depressurization System (ADS) initiating circuits as receiving inputs from eight Reactor Vessel Level transmitters. Design detailing has specified this as four transmitters in each of two separated divisions (total of eight transmitters) that are used to provide the Level 1 initiation signal. The wording is restated in the STP 3 & 4 FSAR as follows:

"All four transmitter signals are fed into the two-out-of-four logic for each of the two divisions, either of which can activate the ADS."

The design as shown in the B21 Logic diagrams as provided in Chapter 21 of the DCD utilizes four level transmitters to measure level 1.

Technical Specification 3.3.1.1 Bases (Applicable Safety Analysis, LCO, and Applicability discussion for Function 9.a, 9.b, and 9.c, Reactor Vessel Water Level - Low, Level 1) is changed to show the Reactor Vessel Water Level - Low, Level 1, originates in four level transmitters, not eight, with these four level transmitters providing reactor vessel water level signals to each of three divisions for these Functions. The certified design specified eight level transmitters, with four transmitters supplying division I and division III, and four transmitters supplying division II.

Evaluation Summary

Special circumstance (iv) applies in that the departure represents a net benefit to the public health and safety by providing a more sophisticated and detailed method of measuring reactor water level and a more robust logic arrangement less prone to spurious actuation.

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

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

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 SRV control switch as “keylock type”
- Manual actuation of ADS by “pushbuttons”

The ADS inhibit and SRV control switches are no longer the keylock type and the ADS manual actuation is now initiated by a single pushbutton. The subsection is modified in the STP 3 & 4 FSAR to present the current design.

Technical Specification 3.3.1.4 Bases (Background) is changed to show the configuration for ADS manual initiation is a single pushbutton (arm and depress) in each division, with each divisional ADS manual initiation switch providing an initiation signal to both the Digital Logic Controllers (DLCs) in that ADS division through a single switch contact. The standard design specified two ADS manual initiation push buttons in each ADS division with each providing an initiation signal to a separate DLC in that ADS division. These Bases are also changed to show that each divisional ADS Manual Inhibit Switch (there is a single switch in each division) provides an initiation signal to both the Digital Logic Controllers (DLCs) in that ADS division through a single switch contact. The standard design specified a single ADS manual inhibit switch in each ADS division with two contacts, with each contact providing an initiation signal to a separate DLC in that ADS division.

Evaluation Summary

Special circumstance (iv) applies in that the departure represents a net benefit to the public health and safety by providing a method evaluated, on net, as more responsive to operator action with fewer mechanical impediments.

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

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

Description

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

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

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

Evaluation Summary

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

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

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

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

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

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

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

Evaluation Summary

Special circumstance (ii) applies in that the departure represents no change in the underlying purpose of the design but clarifies the Technical Specifications and conforms them to the standard design.

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

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

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

- Add variable Type A classification to the Drywell Pressure in Subsection 7.5.2.1. This information is to be used by the control room operator in determining an initiation of the drywell spray to maintain the reinforced concrete containment vessel below temperature limits under a LOCA condition.
- Correct the variable ranges for the Drywell Pressure and Meteorological Data in Table 7.5-2.
- Delete the Secondary Containment Air Temperature from the list of PAM variables.
- Add Wetwell Spray Flow to the list of PAM variables.
- Add a new area radiation instrument to Table 12.3-3 and Figure 12.3-56 for the unmonitored RHR equipment area "C" at Reactor Building Elevation -8200.
- Add the Type A wetwell pressure parameter to the large display panel in Subsection 18.4.2.11 to be used by the control room operator in determining a manual initiation of the wetwell spray to limit the bypass leakage.

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

Technical Specification 3.3.6.1 (Table 3.3.6.1-1) and the associated Bases (LCO discussion for Function 5.a) are changed to show that Drywell Pressure and Wetwell Pressure are Type A Instruments (post-accident monitoring variables), and to show that Wetwell Atmosphere Temperature is a Category I variable and is a required function for post-accident monitoring. The name for Function 5.b in TS Table 3.3.6.1-1 is changed to Wetwell Pressure from Containment Wide Range Pressure, and Wetwell Atmosphere Temperature is added to TS Table 3.3.6.1-1 (Function 13). The Bases are changed to show that Drywell Pressure and Wetwell Pressure are Type A Instruments (post-accident monitoring variables) and show the correct reference to Wetwell Pressure, rather than Containment Wide Range Pressure. The Bases are changed to provide a discussion for Wetwell Atmosphere Temperature (Function 13).

Evaluation Summary

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

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

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

Subsection 7.7.1.2.1 of the reference ABWR DCD provides the Control Rod Drive Control System Interfaces description regarding single rod movement, withdrawal cycle (deleted), insert cycle (deleted), and ganged rod motion (changed to movement). The description was based upon an initial design. Numerous changes have been made in the FSAR:

- The CRT display is replaced with the RCIS Dedicated Operator Interface, a flat panel touch screen.
- A discussion of the RAPI enforcing rod blocks based upon signals external and internal to the system is added.
- Additional discussion in the FSAR regarding the interface of RSM with the synchro position feedback signals adds that the SDCs of the RSM also interface with instrumentation of the FMCRD. Rod position information is collected by the SDCs for the corresponding FMCRD and Synchro A and B analog signals are converted into digital data representing the FMCRD rod position.
- Changes were required for consistency with the final design implementation to reflect the use of soft controls. Also, a clarification regarding no abnormal conditions that prevent establishing automatic rod movement was added for accuracy in the last paragraph of item (2).

Technical Specification 3.9.4 Bases (Action A) is changed to show the alternate method used to ensure a control rod is fully inserted.

Evaluation Summary

Special circumstance (iv) applies in that the departure represents a net benefit to the public health and safety by providing additional interface and feedback check on rod position and control functions. It also better integrates the control rod interface with overall control room human factors design.

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

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

Subsection 7.7.1.2.3 of the reference ABWR DCD is the “Reactor Operator Information” subsection for the RCIS. Current detailed design has resulted in changes to the original listing of alarms, displays and controls. The changes affect the following:

- New annunciation (alarms) for the RCIS - Rod insert block and RWM Trouble
- Status information formerly provided solely on the RCIS Dedicated Operator's Panel (DOI) - not all of the status information as described is available on the RCIS DOI itself. Some status is information is now shown on MCRP display (e.g. Wide Panel display) or other locations on the MCPR where it can be visible to more than the one operator seated at the DOI.
- Logic and control actions available on the Dedicated Operator's Panel - clarifying wording is provided because not all of the control actions listed are available at the RCIS DOI. Some of the related controls are available on MCRP itself and some are at RCIS related panels.
- Logic and control actions available on the Dedicated Operator's Panel - not all of the control actions listed are available at the RCIS DOI. Some of the related controls are now available on MCRP itself and some are at RCIS related panels and are no longer exclusively accessible from the DOI.
- Information was to be displayed on CRT displays but current design utilizes flat panel displays. Therefore, the subsection is changed to reflect the latest ABWR main control room panel display design details.

Technical Specification 3.9.3, 3.10.4, and 3.10.5 Bases (Background) and 3.10.3 Bases (Background, and Applicable Safety Analysis) are changed to show the change to a specific mode that the RCIS is placed into (i.e., scram test mode) rather than specify a component that initiates the mode.

Evaluation Summary

Special circumstance (iv) applies in that the departure represents a net benefit to the public health and safety by more effectively integrating the RCIS functions with the overall control room human factors design and with other operator information presentation devices.

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

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

Licensing Topical Report NEDO-33335, "Plant Medium Voltage Electrical System Design," was submitted to the NRC by General Electric Company in May 2007, proposing this change as a generic revision to the reference ABWR DCD. The LTR is incorporated by reference in FSAR Sections 8.1, 8.2, 8.3, and Appendix 8A.

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

- Medium voltage rating of the Class 1E buses to 4.16 kV
- EDG ratings to 7200 kW and 4.16 kV,
- Combustion turbine generator (CTG) voltage rating to 13.8 kV, and
- Time required for CTG to start and achieve steady state voltage and frequency from two minutes to "less than 10 minutes"

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

10 CFR 50.63 requires the CTG (alternate AC source) to be available to power the Class 1E buses within 10 minutes of the onset of an SBO. The change in CTG startup time from two to < 10 minutes does not affect plant safety in the station blackout event since the CTG is not required to provide immediate core inventory or reactor pressure control.

As a result of the LTR and its incorporation into the STP 3 & 4 design, Technical Specifications 3.3.1.4 (Table 3.3.1.4-1), and Bases 3.3.1.1 (Background) and 3.3.1.4 (Applicable Safety Analysis, LCO, and Applicability, Actions) are changed to show the medium voltage is 4.16 kV, not 6.9 kV.

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

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

Evaluation Summary

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

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

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

STP DEP 10.4-5, Condensate and Feedwater System

The reference ABWR DCD and the STP 3 & 4 design are compared in the table below:

Reference ABWR DCD	STP 3 & 4
■ Four condensate pumps ■ Three reactor feed pumps ■ Two heater drain pumps ■ One or more heater drain tanks ■ A feed pump bypass valve controls FW during plant startup	■ Four condensate pumps ■ Four condensate booster pumps ■ Four reactor feed pumps ■ Four heater drain pumps ■ One heater drain tank ■ One low flow control valve in feed pump discharge header for startup ■ One bypass valve used for bypassing HP heaters

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

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

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

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

Evaluation Summary

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

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

2.2.2 Changes of Intent to the Technical Specifications

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

STD DEP 16.2-1, Safety Limit Violation

Description

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

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

Specifications 2.2.1 and 2.2.4 are duplicative of the requirements found in 10 CFR 50.36.c. (1) and 10 CFR 50.73 respectively; Specifications 2.2.3 and 2.2.5 do not meet the criteria for inclusion in Technical Specifications; 2.2.3 will be relocated to a plant specific document controlled by 10 CFR 50.59 and 10 CFR 50.36 (c)(1) requires unit shutdown on a Safety Limit violation. Specification 5.5.2 is not a valid specification.

Evaluation Summary

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

- The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the actions in question are required by regulations or plant documents and therefore the deletions of the actions from the Technical Specifications will not present an undue risk to the public health and safety, and the departure does not relate to security and does not otherwise pertain to the common defense and security.
- Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since it is unnecessary to include the deleted provisions in the Technical specifications in order to ensure that the actions are accomplished.

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

References

(1) 10 CFR 50.36, Technical Specifications

(2) 10 CFR 50.12, Specific Exemptions

STD DEP 16.3-38, LCO 3.3.4.1, SSLC Sensor Instrumentation

Technical Specification 3.3.4.1 and its associated bases are changed based upon the ABWR design for this instrumentation. Specifically:

Condition A is revised to remove the bypass Action and the trip Action since these features are not available for these Functions.

New Action B.2 is added for Function 14, which was previously addressed in Action B.1, since this Function only has two channels, not three as shown for Functions 1, 3, 5, and 11. The note for Action B.1 is revised, and a new note for Action B.2 is added, to differentiate between the Functions and the applicable action.

Condition C note is revised to show that Condition C Actions only apply to Function 2 and 4, and not to Function 9. Function 9 does not have bypass or trip capability so these Actions (C.1.1, C.1.2.2, and C.2) do not apply to Function 9. A new Condition D is added to provide the Action for Function 9. The subsequent Conditions (D, E, F, G, and H) were renumbered based upon the Addition of new Condition D.

Action H.1, which is shown as Action I.1 in the markup because of renumbering as described above, is revised to remove the associated RIP form service, rather than just declare the supported feature (the associated RIP) inoperable. An appropriate time for removing the RIP from service is also provided for the revised Action.

In Table 3.3.4.1-1, Function 9 is revised to remove SR 3.3.4.1.2 as applicable to this Function, since testing of this Function requires the plant to be in a MODE where the Function is not required before it can be tested. SR 3.3.4.1.4, LOGIC SYTEM FUNCTIONAL TEST, and SR 3.3.4.1.6, COMPREHENSIVE FUNCTIONAL TEST, are both performed at the refueling interval (18 month interval) and provide the necessary testing in a Mode where this Function is not required.

In Table 3.3.4.1-1, Function 14 is revised to show the Required Channels as 2 (not 3) since this Function only has two channels. As described previously, this change in the number of Required Channels has been addressed in the changes to the Actions Table.

Table 3.3.4.1-1 footnote (a) is updated to show the RIPs that have timers.

The Bases for this specification have been changed because this specification changed. In addition to changes to the Bases that are representative of the changes to the specification, the following Bases changes are made based upon the ABWR design for this instrumentation.

In addition, the Descriptions in the Applicable Safety Analysis for the following Functions are updated: Function 1, Feed Reactor Vessel Water Level - Low, level 3; Function 3, SB&PC Reactor Steam Dome Pressure - High; Function 4, EOC-RPT Instrumentation; Function 6, Adjustable Speed Drive Pump Trip Actuation; Functions 7 and 8, Adjustable Speed Drive Pump Trip Timers and Interrupters; Function 9, RPS Scram Follow Signal; Function 14, ATWS-ARI Valve Actuation; and Function 16, Recirculation Runback.

Evaluation Summary

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

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

- Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since these changes in the Technical specifications are required to accurately reflect the DCD design descriptions in Tier 1 and Tier 2.

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

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

TS 3.3.4.2 was updated consistent with the Feedwater Pump and Main Turbine Trip Instrumentation design, which uses the fault tolerant digital controllers. LCO Statement, ACTIONS, Surveillance Requirements, and associated Bases have been updated to reflect the actual ABWR design.

Evaluation Summary

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

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

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

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

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

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

Evaluation Summary

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

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

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

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

STD DEP 16.3-79, LCO 3.3.1.1, Safety System Logic and Control (SSLC) Sensor Instrumentation

TS 3.3.1.1, Table 3.3.1.1-1 Surveillance Requirements are updated based upon the ABWR instrumentation design. The specific changes are as described below:

TS Table 3.3.1.1-1 the SRs listed for Function 1.b, SRNM Neutron Flux-Short Period, for Applicable Mode 2 does not include the CoFT SR 3.3.1.1.9 or the Sensor Channel Calibration SR 3.3.1.1.10, however, both of these are listed for this Function as Applicable in Mode 5. These SRs are added to the Surveillance Requirements column for Function 1.b during Applicable Mode 2. This change is consistent with the SRs listed for this Function for Mode 5.

Table 3.3.1.1-1 Function 1.d, SRNM-Inop, for Applicable Modes 1 and 2 does not include the CoFT SR 3.3.1.1.9. This SR is added to the Surveillance Requirements column for Function 1.d during Applicable Modes 1 and 2. This change is consistent with the SR listed for this Function for Mode 5.

Table 3.3.1.1-1 Function 2.a, APRM Neutron Flux - High, Setdown: SR 3.3.1.1.9 and SR 3.3.1.1.10 are not listed but are required SRs for this Function. These SRs are added to the Surveillance Requirements column for Function 1.b during Applicable Mode 2. This change is consistent with the SRs listed for other APRM Functions (2.b, 2.c, 2.e).

Table 3.3.1.1-1 Function 2.d, APRM-Inop: SR 3.3.1.1.9 is not listed but is a required SR for this Function. This SR is added to the Surveillance Requirements column for Function 2.d during Applicable Mode 2. This change is consistent with the SRs listed for other APRM Functions (2.b, 2.c, 2.e).

Table 3.3.1.1-1 Function 2.f, Oscillation Power Range Monitor: Removed SR 3.3.1.1.1 as the Sensor Channel checks do not apply to the OPRM. Added SR 3.3.1.1.7 (LPRM cal) since the OPRM depends on signals from the LPRMs to perform its function. This change is consistent with the SRs shown for APRM Functions, which also depend on signals from the LPRMs to perform their function.

Table 3.3.1.1-1 Function 12, CRD Water Header Charging Pressure-Low, for Applicable Modes 1 and 2: SR 3.3.1.1.9 and SR 3.3.1.1.10 are not listed but are required SRs for this Function. These SRs are added to the Surveillance Requirements column for Function 12 during Applicable Modes 1 and 2. This change is consistent with the SRs listed for this Function for Mode 5.

Table 3.3.1.1-1 Function 24.a, Reactor Building Area Exhaust Air Radiation-High, for Applicable Modes 1, 2, 3: SR 3.3.1.1.9, SR 3.3.1.1.10, and SR 3.3.1.1.14 are not listed but are required SRs for this Function. These SRs are added to the Surveillance Requirements column for Function 24.a during Applicable Modes 1, 2, and 3. This change is consistent with the SRs listed for this Function for Mode 5.

Table 3.3.1.1-1 Function 24.b, Fuel Handling Area Exhaust Air Radiation-High, for Applicable Modes 1, 2, 3: SR 3.3.1.1.9, SR 3.3.1.1.10, and SR 3.3.1.1.14 are not listed but are required SRs for this Function. These SRs are added to the Surveillance Requirements column for Function 24.a during Applicable Modes 1, 2, and 3. This change is consistent with the SRs listed for this Function for Mode 5. This change is consistent with the SRs listed for this Function for Mode 5.

Evaluation Summary

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

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

(2) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since the departure represents a clarification of otherwise conflicting wordings and descriptions within the Technical Specifications and to the Tier 1 and Tier 2 designs.

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

STD DEP 16.5-1, Unit Responsibility

Description

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

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

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

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

Evaluation Summary

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

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

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

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

References

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

STD DEP 16.5-3, Technical Specification Bases Control Program

Description

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

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

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

Evaluation Summary

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

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

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

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

STD DEP 16.5-4, Reporting Requirements

Description

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

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

10 CFR 20.2066 requires this due date by April 30 of each year, this change is necessary to be consistent with the CFR.

Evaluation Summary

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

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

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

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

References

(1) 10 CFR 20.2206 Reports of Individual Monitoring

2.2.3 Technical Specifications Editorial Revisions and Clarifications

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

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

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

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

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

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

STD DEP 16.2-2, Safety Limits

Description

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

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

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

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

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

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

In addition, the Bases for SR 3.0.1 refers to SR 3.1.3.4 for scram time testing. The appropriate SR is SR 3.1.4.3. SR 3.1.4.3 is the Surveillance performed at lower reactor pressures and is appropriate for this Bases discussion.

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

The Bases states that "Because the minimum required boron solution concentration is the same for both ATWS mitigation and cold shutdown (unlike some previous reactor designs) then if the boron solution concentration is less than the required limit, both SLC subsystems shall be declared inoperable." The Required Actions for LCO 3.1.7 include Condition A that requires entry when concentration of boron in solution is not within limits. The Required Action is to restore concentration of boron in solution to be within limits within 72 hours and 10 days from discovery of failure to meet the LCO. The Bases is not consistent with the LCO therefore, the Bases statement has been deleted.

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

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

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

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

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

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

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

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

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

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

The Bases Applicable Safety Analyses of LCO 3.4.9 states "Reference 7 establishes the methodology for determining the P/T limits." Reference 7 is NEDO 21778-A. Specification 5.7.1.6 includes a bracketed place holder so that a COL applicant will provide all the analytical methods used to determine the pressure and temperature limits and the heatup and cooldown rates. NEDO 21778-A is not the correct reference. This document does not include the methodology for determining P/T Limits for the ABWR. Therefore it has been deleted as a reference and a bracketed requirement has been added to provide the "ABWR P/T Limit Methodology." This change corrects the LCO 3.4.9 Bases making it consistent with Technical Specification 5.7.1.6. A new methodology will be developed prior to fuel load.

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

The Bases required action for LCO 3.4.7 and LCO 3.4.8 states that the Spent Fuel Pool Cooling System may be used as an alternate decay heat removal system during MODE 3 and MODE 4, respectively. The Bases of LCO 3.9.7 and LCO 3.9.8 does not include the Spent Fuel Pool Cooling System as a method for alternate decay heat removal system during MODE 5 operations. The Spent Fuel Pool Cooling System cannot be used for decay heat removal in MODE 3 and 4 so it has been deleted from Specifications 3.4.7 and 3.4.8, however it can be used in MODE 5 therefore, it has been added to the Bases of 3.9.7 and 3.9.8.

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

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

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

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

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

STD DEP 16.3-11, 3.4.3 RCS Operational LEAKAGE

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

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

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

The Bases describe the flow path of the RHR Shutdown Cooling System to the reactor pressure vessel. For RHR subsystems B and C it currently states that each pump discharge to the reactor is via the "RHR inlet nozzles." This has been changed to "RHR low pressure flooder spargers" consistent with the Bases of 3.4.7.

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

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

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

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

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

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

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

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

LCO 3.7.1 includes a Required Action C.1 that requires restoration of two inoperable RCW/RSW or UHS divisions to OPERABLE status within 14 days. LCO 3.7.2 includes a Required Action B.2 that requires restoration of two inoperable RCW/RSW or UHS divisions to OPERABLE status within 14 days. These Required Actions have been deleted since redundant requirements are included in other Condition A of each Specification. The change is consistent with the Completion Time Rules of Section 1.3.

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

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

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

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

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

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

LCO 3.10.8 is applicable in MODE 5 with the reactor mode switch in startup/hot standby position. The Bases Applicability section states, “These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned.”

The Bases has been modified to fully reflect the applicability consistent with the Specification. MODE 5 is already defined in Table 1.1-1 with the reactor mode switch in the “Shutdown or Refuel” position and with one or more reactor vessel head closure bolts less than fully tensioned. It is clear from the definition that when the head is removed, the plant is in MODE 5 when the reactor mode switch is in the Shutdown or Refuel position. The important detail is the exception reflected in the LCO Applicability (i.e., the reactor mode switch position is in the startup position.

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

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

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

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

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

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

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

Technical Specification LCO 3.3.1.1, Functions 2.a, APRM Neutron Flux-High, Setdown and Function 2.d, APRM-Inop is applicable in MODES 2 and MODES 1 and 2, respectively. LCO 3.10.5 is applicable in MODE 5 with LCO 3.9.5 not met. LCO 3.10.5 allows the requirements of and Function 2.a and Function 2.d to not me met when in utilizing this Special Operations LCO. Since LCO 3.10.5 is used when in MODE 5 with LCO 3.9.5 not met, there is no specific need to except the requirements of Function 2.a and 2.b. Therefore they have been deleted from the LCO statement.

STD DEP 16.3-22, LCO 3.10.5, Control Rod Drive (CRD) Removal - Refueling

LCO 3.10.5 allows the removal of a single CRD or CRD pair associated with control rod(s) withdrawn from core cell(s) containing one or more fuel assemblies, provided certain requirements is met. SR 3.10.5.1 and SR 3.10.5.2 imply only one control rod can be removed. They have been modified to be consistent with the LCO allowance. Similar changes have been made in the Bases for Required Action A.1, A.2.1, and A.2.2 and the Bases descriptions of SR 3.10.5.1, SR 3.10.5.2, SR 3.10.5.3, SR 3.10.5.4, and SR 3.10.5.5.

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

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

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

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

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

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

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

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

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

The Bases Background of LCO 3.10.2 discusses the reactor mode switch positions and the related scram interlock functions. The list included requirements on reactor high water level. The ABWR does not include a “reactor high water level trip.” Therefore, the Bases have been modified to be consistent with the DCD.

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

LCO Bases provides a list of other Special Operations LCO applicable in MODE 3, 4, 5 operations. In this list, "LCO 3.10.7, Control Rod Testing - Operating," is listed. This Specification is applicable in MODE 1 and 2 with LCO 3.1.6 not met. Reference to the Specification is deleted since it does not apply. However, additional Specifications have been added that are applicable in MODE 3, 4, and 5.

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

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

STD DEP 16.3-29, LCO 3.6.4.1, Secondary Containment

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

STD DEP 16.3-30, LCO 3.6.4.1, Secondary Containment

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

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

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

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

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

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

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

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

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

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

The Bases ACTIONS have been changed accordingly.

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

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

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

The LCO Bases states "All eight of the vacuum breakers must be OPERABLE for opening. All wetwell-to-drywell vacuum breakers, however, are required to be closed (except during testing or when the vacuum breakers are performing the intended design function)."

Each vacuum breaker is verified closed (except when being tested in accordance with SR 3.6.1.6.2 or when performing its intended function) to ensure that this potential large bypass leakage path is not present.

The allowance in the Bases that the wetwell-to-drywell vacuum breakers may be opened during testing has been deleted since it is not stated in the LCO or SRs. This is acceptable since there are no requirements to perform actuation tests during the operating cycle.

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

The Background bases states that "Sufficient iodine activity would be retained to limit offsite does from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3." The limit has been changed to "< 25% of 10 CFR 100 limits." This limit is consistent with Reference 3, NUREG-0800, Section 15.7.4.

The Applicable Safety Analysis states "A minimum water level of 7.0 m allows a decontamination factor of 100 (Ref. 4) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water." Reference 4 is "NUREG-0831, Supplement 6, Section 16.4.2." This reference has been deleted and Reference 4 in the above statement has been changed to Reference 1 (Regulatory Guide 1.25, March 23, 1972). Regulatory Guide 1.25 is the appropriate reference for the decontamination factor. In addition, the word "dropped" has been changed to "damaged." This is consistent with the analysis (DCD Section 15.7).

The Applicable Safety Analyses section states, "With a minimum water level of 7.0 m and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite does are maintained within allowable limits (Ref. 5)." Reference 5 has been renumbered as Ref. 4.

The Applicability states "Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.6, "Fuel Pool Water Level." The Specification number has been changed from 3.7.6 to 3.7.8. Specification 3.7.8 is the appropriate number.

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

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

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

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

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

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

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

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

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

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

STD DEP 16.3-43, LCO 3.6.1.1, Primary Containment

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

STD DEP 16.3-44, LCO 3.6.1.1, Primary Containment

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

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

STD DEP 16.3-45, LCO 3.6.1.1, Primary Containment

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

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

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

The Applicability of 3.7.2 is in MODE 5 except with the reactor cavity to dryer/separator storage pool gate removed and water level ≥ 7.0 m over the top of the reactor pressure vessel flange. The Applicability of LCO 3.7.3 is MODE 5 with the reactor cavity to dryer/separator storage pool gate removed and water level > 7.0 m over the top of the reactor pressure vessel flange. The Applicability requirements of these Specifications should match the Applicability of Specifications 3.9.7 and 3.9.8. The applicability of Specification 3.9.7 is MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level ≥ 7.0 m above the top of the RPV flange while the Applicability of LCO 3.9.8 is MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level < 7.0 m above the top of the RPV flange. The Applicability of Specifications 3.7.2 and 3.7.3 has been changed to be consistent with the Applicability of 3.9.8 while the Applicability of 3.7.3 has been changed to be consistent with the Applicability of 3.9.7. The Bases Background, Applicable Safety Analyses, LCO, and Applicability have been corrected.

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

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

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

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

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

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

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

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

STD DEP 16.3-51, LCO 3.8.2, AC-Sources-Operating

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

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

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

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

STD DEP 16.3-53, LCO 3.3.1.1, SSLC Sensor Instrumentation

In SR 3.3.1.1.14, ISOLATION SYSTEM RESPONSE TIME acceptance criteria, are moved from Reference 10 to REFERENCE 9 and in the REFERENCE Section, Reference 9 is "Technical Requirements Manual" and REFERENCE 10 is not used. The changes are to universally locate all RESPONSE TIME acceptance criteria in the Technical Requirements Manual. SR 3.3.1.12 specifies Reference 9 as the location for Response Time Tests. Reference 9 changed to Technical Requirements Manual.

STD DEP 16.3-54, LCO 3.3.1.2, Reactor Protection System (RPS) and Main Steam Isolation Valve (MSIV) Actuation

The phrase, "Response time test acceptance criteria are included in Reference 4." is added to SR 3.3.1.2.6 and SR 3.3.1.2.7. "Technical Requirements Manual" is added to the REFERENCES section Item 4.

These additions, regarding response time acceptance criteria, provide one location for all response time criteria.

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

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

STD DEP 16.3-56, LCO 3.3.4.1, Anticipated Transient Without Scram (ATWS) and End-of-Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

Reference 5 of the LCO 3.3.4.1 Bases has been changed to "Technical Requirements Manual."

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

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

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

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

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

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

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

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

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

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

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

STD DEP 16.3-62, LCO 3.3.8.2, Electric Power Monitoring

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

The associated Bases only refers to Condition B. Therefore, Condition A has been added to the Bases description.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

This change results in a number of changes:

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

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

ABWR DCD SR 3.6.1.3.13 specifies leakage rate limits for the main steam isolation valves. This SR contains a Note that states the results shall be evaluated against acceptance criteria of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. This Note has been deleted since the Containment Radiological Analysis takes into account MSIV leakage in addition to La. Also, this is acceptable if main steam line leakage is not within limit ACTION D must be entered and only a short time is allowed to restore leakage before a shutdown action is entered. The corresponding changes are also made to the Bases of 3.6.1.1 and the Bases of SR 3.6.1.3.2.

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

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

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

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

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

The SRs applicable in the Specification cover purge valves and MSIVs. These valves are secondary containment bypass penetrations. There are no other SRs associated with secondary containment bypass penetrations. The Bases states, “Purge valves with resilient seals, secondary bypass valves, MSIVs, and hydrostatically tested valves must meet additional leakage rate requirements.” The secondary bypass valves have been deleted from the phrase since there are no additional requirements for secondary bypass valve leakage.

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

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

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

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

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

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

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

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

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

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

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

The Applicable Modes or Other Specified Conditions for SRNM and APRM LOGIC CHANNELS (Function 1a) has been added as footnote (b) to Table 3.3.1.2-1. The footnote appropriately requires the SRNM and APRM LOGIC CHANNELS to be OPERABLE when the associated Function in LCO 3.3.1.1 is required to be OPERABLE. The bases discussion in the APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY has also changed based upon the change to the specification.

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

The bases discussion for Required Action B.1, B.2, and B.3 and Required Action F.1 and F.2 is changed to specify that the two inoperable channels are “for the same Function.”

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

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

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

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

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

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

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

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

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

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

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

STD DEP 16.3-88, LCO 3.3.1.4, ESF Actuation Instrumentation

Change REFERENCE 5 to “Technical Requirements Manual”. Response time testing acceptance criteria are universally to be shown in this manual and not in DCD 1.1.3.

STD DEP 16.3-89, LCO 3.1.2, Reactivity Anomalies

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

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

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

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

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

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

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

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

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

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

The Applicable Modes or Other Specified Conditions for Function 13c, CUW Isolation and SLC Initiation, in Table 3.3.1.4-1 is changed from MODE 1, MODE 2, and MODE 3, to only MODE 1 and MODE 2, since these are the MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC system. Events involving control rods are presented in References 2, 3, 4, and 5. Reference 5 is the rod ejection accident. In accordance with Reference 5 the event is not postulated to occur, therefore it has been deleted.

STD DEP 16.5-2, Unit Staff

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

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