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.

~~STD 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 2.1-2, Reactor Pressure Vessal System RIP Motor Casing Cladding

Description

This requested departure modifies the description of the RIP motor casing to clearly indicate that some portions of the motor casing have cladding.

ABWR DCD Tier 1 describes the cladding applied to the interior of the RPV cylindrical shell and indentifies areas of the RPV where there is no cladding. Specifically, DCD Tier 1 Section 2.1.1 states that there is no cladding on the RIP motor casing. The standard ABWR design for installed applications includes stainless steel cladding from the top portion of the casing to the motor secondary seal (along the length of the stretch tube) and around the bottom of the RIP motor casing where it interfaces with the motor cover closure. The requested departure modifies the DCD Tier 1 RIP motor casing design description to be consistent with the ABWR RIP motor casing design in current use. The design description in the ABWR DCD Tier 2 Section 5.3.3.1.1.1 is also clarified for consistency with Tier 1.

Evaluation Summary

This departure changes the RIP motor casing to incorporate cladding in the stretch-tube portion above the RIP secondary seal and at the bottom end of the casing near the closure of the motor cover.

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 provides protection of the RIP motor casing base metal, and as such is an enhancement to the design that 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 enhancement 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 circumstances (iv) is present, since the design change represents an improvement based on ABWR operating experience. The change is proven effective by operating history 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 does not reduce safety, and does not affect the configuration of the plant or the manner in which the plant is operated. Further, this departure is consistent with operatiing ABWR designs, and will form the reference-COLA for future COL applicants, thus the departure will likely not affect standardization. Any reduction in standardization resulting from the change in the RIP motor casing cladding will 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 nonsafety-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 nonsafety-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 control 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 18F-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.~~

The Scram and MSIV Automatic Closure on high MSLRM (main steam line radiation monitor) trip is deleted. This safety function is deleted for the following reasons:

- The MSLR-high trip is not specifically credited in any ABWR safety analysis. This trip was originally designed for BWRs to mitigate the effects of a control rod drop accident (CRDA). As described in Tier 2 DCD Section 15.4.10, the ABWR has no basis for the CRDA event to occur. Thus, the deletion of the automatic scram and MSL isolation results in no change in associated risk or safety margins.
- U.S. BWRs have experienced spurious trips due to this function. The radiation trip setpoints can be overwhelmed by minor variations in the N-16 flow during normal operation and cause spurious trips. Elimination of the safety-related functions reduces the potential for unnecessary reactor shutdown and increases plant operational flexibility. Operators in the main control room are alerted to potential offsite releases by the MSLRM, the condenser steam jet air ejector monitor, and/or ventilation stack monitor.

Furthermore, this change has been previously approved by the NRC for U.S. BWRs based on analyses that demonstrate that safety margins are not impacted. Since the SER conditions are met for the ABWR, as explained above, no other safety analyses are required.

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

Tier 1 Departures:

- ABWR Tier 1 DCD Figure 2.3.1, “Process Radiation Monitoring System Control Interface Diagram” is changed to remove the MSL Tunnel Area Radiation input from the plant sensors that provide input data.
- Tier 1 Table 2.7.1a has been modified to remove the main steam tunnel radiation information from the fixed position alarms, displays, and controls. This information is conveyed through other alarms, displays, and controls in the control room.

Technical Specifications Departures:

- LCO 3.3.1.1 and its associated Bases have been modified to remove the Main Steam Tunnel Radiation High functions (automatic scram and MSIV closure).
- LCO 3.3.6.1 and its associated Bases have been modified to remove instrumentation monitoring functions for post-accident monitoring (PAM) of coolant radiation in the main steamline. A continuous PAM for coolant radiation is no longer required based on BTP HICB-10.

Tier 2 Departures:

Changes have been made relative to the reference ABWR Tier 2 DCD Sections 1.2, 1A, 3.4, 5.2, 6.2, 7.1, 7.2, 7.3, 7.5, 7.6 11.5, 15.2, 18F, and 18H to revise or remove information pertaining to main steam line high radiation monitoring and process radiation monitoring system. For example, Section 11.5 is 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.”

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. This departure revises ABWR Tier 1, Sections 2.4.3 and 2.15, and the Tier 2 sections, including Technical Specifications, affected by the revision.

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, as discussed in DCD Tier 2, Subsections 6.2.1.1.3.3.1.2 and 6.2.1.1.5.6.1. With the current ABWR design, the only mitigation option available, for limiting the containment pressure, would be operator action using the non-safety trip of the condensate and/or feedwater pumps.

Therefore, high drywell pressure signals that would already be existing in the Leak Detection & Isolation ~~(LDI)~~ (LDS) logic of the Safety System Logic & Control (SSLC) are used, in conjunction with the added 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. This is described in Departure 6.2-2, Containment Analysis (see Departures from the General Technical Specifications) and Tier 2, Subsection 6.2.1.1.3.3.1. 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~~ these 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. The design and location of the safety related breakers are described in Tier 2, Subsection 8.3.1.1.1.

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) monoblock design (pump and 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 monoblock 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. This design has been installed and is operational in international nuclear and fossil power plants as well as in maritime and military applications.

The Tier 2 impacts follow from design simplification and design classification upgrades. Changes are made to the Tier 2 mechanical, control, and testing sections. 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 pump and turbine can now be fabricated to ASME Section 3 requirements. 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~~ The containment penetration for the RCIC vacuum pump discharge line ~~and upgraded to burst pressure have~~has 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.

The ITAAC in 2.4.4 (c), (e), and (f) are modified to reflect the fact that the steam supply bypass valve is not used for startup and a 10-second time delay is no longer needed for the injection and steam admission valves. Also, the ITAAC 2.4.4 (i) (2) associated with the torque to the pump is deleted because of the monoblock design.

Technical Specification Table 3.3.1.4-1, ESF Actuation Instrumentation and SSLC Sensor Instrumentation item 12 d is reinstated and “RCIC Turbine exhaust diaphragm pressure” is corrected to "RCIC turbine exhaust pressure" in this item and in the bases.

~~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.~~ A correction is made to the RCIC system performance test discussion in Subsection 12.2.12.1.9(3)(f)(iv) to clarify that the test return line discharges to the suppression pool and not the condensate storage tank.

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~~ will have a positive effect on plant safety as ~~will improve~~ well as transient and startup characteristics. Changes to the RCIC ITAAC are simplications due to fewer components yet still allow demonstration of performance critical to safety.

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 nonsafety-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”~~

Description

10 CFR 50.44, "Combustible gas control for nuclear power reactors," was amended after the issuance of the design certification for the ABWR. The amended 10 CFR 50.44 eliminates the requirements for hydrogen control systems to mitigate a design-basis LOCA hydrogen release. As a result of this change, the use of the containment hydrogen and oxygen monitoring instrumentation in the mitigation of a design-basis LOCA is also eliminated.

This departure reflects the elimination of the requirement to maintain equipment needed to mitigate a design-basis LOCA hydrogen release. This departure includes the following:

(1) The ABWR Flammability Control System (FCS), which consists of two redundant hydrogen recombiners, is no longer required in the response to a design basis LOCA and is eliminated. In conjunction with this change, LCO 3.6.3.1, "Primary Containment Hydrogen Recombiners," which established the requirements for the FCS is deleted. LCO 3.3.6.2, "Remote Shutdown System," is modified to delete Function 17, which required remote shutdown system controls for cooling water to the FCS. Supports systems associated with the FCS are modified or deleted, as necessary, to support removal of the FCS.

(2) The containment hydrogen and oxygen monitoring functions of the Containment Monitoring System are no longer required to function for the mitigation of a design basis LOCA. Consequently, the containment hydrogen and oxygen monitoring functions are no longer classified as Category 1, as defined in Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 4. The RG 1.97 classification of containment hydrogen and oxygen monitoring functions are changed to Category 3 for hydrogen monitoring, and Category 2 for oxygen monitoring, allowing these instruments to be re-classified as nonsafety-related. In conjunction with this change, LCO 3.3.6.1, "Post Accident Monitoring (PAM) Instrumentation," is modified to delete Functions 11 and 12, requirements for the H_2 and O_2 analyzers in the containment drywell and wetwell. This change to LCO 3.3.6.1 is acceptable because only Category 1 PAM instruments meet 10 CFR 50.36 criteria for inclusion in technical specifications.

With the adoption of these changes, the design and other requirements for control of combustible gases satisfy the regulations in 10 CFR 50.44(c) as amended. The design and requirements for control of combustible gases are consistent with the guidance provided in Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment, Revision 3, dated March 2007, as described below.

(1) 10 CFR 50.44(c)(1), Mixed atmosphere, requires that all containments have a capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents. Section C.3 of RG 1.7 specifies

that this capability may be provided by an active, passive, or combination system. Active systems may consist of a fan, a fan cooler, or containment spray.

The ABWR satisfies this requirement by a combination of active and passive capability. As indicated in the reference ABWR DCD, Section 6.2.5.1(6), the drywell and the suppression chamber will be mixed uniformly after the design basis LOCA due to natural convection and molecular diffusion. Mixing will be further promoted by operation of the containment sprays. The containment spray system consists of two RHR spray loops, each of which includes both wetwell and drywell sprays. LCO 3.6.2.4, Residual Heat Removal (RHR) Containment Spray," ensures that the active components for containment mixing are reliable, redundant, single-failure-proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power as recommended in Section C.3 of RG 1.7.

(2) 10 CFR 50.44(c)(2), Combustible gas control, requires that all containments have an inerted atmosphere or must limit hydrogen concentrations in containment during and following an accident.

The ABWR satisfies this requirement with the Atmospheric Control System (ACS), which is provided to establish and maintain an inert atmosphere within the primary containment. LCO 3.6.3.2, "Primary Containment Oxygen Concentration," ensures that the primary containment is inerted whenever reactor power is greater than 15% of rated thermal power.

(3) 10 CFR 50.44(c)(3), Equipment Survivability, requires that containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen.

This requirement is not applicable to the ABWR because the ABWR uses an inerted atmosphere to control combustible gases in the primary containment.

(4) 10 CFR 50.44(c)(4), Monitoring, requires that equipment be provided for monitoring oxygen and hydrogen in the containment. This oxygen and hydrogen monitoring equipment must be must be functional, reliable, and capable of continuously measuring the concentration of oxygen and hydrogen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

The ABWR satisfies this requirement for monitoring oxygen and hydrogen in the primary containment as described in item (k) of the reference ABWR DCD, Section 7.5.2.1, "Post Accident Monitoring System," and Table 7.5-2, "ABWR PAM Variable List," as modified by this departure. Specifically, the containment hydrogen and oxygen monitoring functions are no longer required to function for the mitigation of a design basis LOCA and are no longer classified as Category 1, as defined in RG 1.97. The oxygen and hydrogen monitors for the containment drywell and wetwell satisfy design requirements consistent with their RG 1.97 classification as Type C, Category 2 (oxygen) and Category 3 (hydrogen) instruments.

(5) 10 CFR 50.44(c)(5), Structural analysis, requires that an applicant perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC. Section C.5 of RG 1.7 specifies that that an acceptable method for demonstrating that these requirements are met for steel containments is conformance with ASME Boiler and Pressure Vessel Code (edition and addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, considering pressure and dead load alone (evaluation of instability is not required). Section C.5 of RG 1.7 further specifies that, as a minimum, the specific code requirements set forth should be met for a combination of dead load and an internal pressure of 45 psig.

The ABWR satisfies this requirement as indicated in ABWR DCD, Section 3.8.2.5, “Structural Acceptance Criteria,” Section 19A.2.45, Containment Integrity [Item (3)(v)], and Section 19E.2.3.2, “100% Metal-Water Reaction.” These sections provide a detailed description of how the ABWR containment satisfies the requirements of 10 CFR 50.44(c)(5) using methods determined acceptable in Section C.5 of RG 1.7.

Evaluation Summary

This evaluation covered Tier 1 and Tier 2 departures .

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) This design change incorporates changes to regulations that occurred after the issuance of the design certification for the ABWR. After incorporation of these design changes, the ABWR ~~DCD~~ design features and requirements for control of combustible gases will satisfy the regulations in 10 CFR 50.44(c) (Ref. 1), consistent with the guidance provided in Regulatory Guide 1.7 (Ref. 2). Therefore, this change 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

STD DEP T1 2.15-1, Re-classification of Radwaste Building Substructure from Seismic Category I to Non-Seismic

Description

The reference ABWR DCD Section 2.15.13 states that the exterior walls of the RW/B below grade and the basemat are classified as Seismic Category I. This departure revises the seismic category of the RW/B substructure from Seismic Category I to non-seismic. The RW/B does not house any safety related systems or components. Regulatory Guide 1.29, Seismic Design Classification, provides a list of SSCs which have to be classified as Seismic Category I. Item p on Page 4 of the Reg. Guide says “systems, other than radioactive waste management systems, not covered by ---“, shall be Seismic Category I. The phrase ‘other than radioactive waste management systems’ excludes these systems from the list of Seismic Category I SSCs. For the radioactive waste management system, the Reg. Guide 1.29 refers to the Reg. Guide 1.143 in Note 5. The detailed guidance for the design of the radwaste processing systems, structures, and components is provided in Regulatory Guide 1.143. This departure commits to follow the guidance of Regulatory Guide 1.143.

Also, NUREG-1503, Section 3.8.4 states that Radwaste Building is not Seismic Category I. The NRC included this design in their review because GE elected to design the RW/B substructure as Seismic Category I.

Based on this departure, the COLA is revised to delete the description and results of RW/B analysis and design from those sections of the COLA which included such description because the RW/B substructure was classified as Seismic Category I structure. Examples of these deleted sections include Sections 2.5S.4, 3.7, 3.8, and Appendix 3H.3. Also, revisions have been made throughout the COLA to appropriately change the seismic classification of the RW/B (Part 7, Table 5.0-1).

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 because the classification of the Radwaste Building in the reference DCD is unduly conservative and is not necessary to satisfy applicable NRC regulations or guidance.

(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.

STD DEP T1 2.15-2 RBSRDG HVAC

Description

ABWR DCD Tier 1 Subsection 2.15.5, “Heating, Ventilating and Air Conditioning Systems” describes the operation and setting of the R/B Safety-Related DG HVAC System to control temperature in the diesel generator (DG) engine rooms during DG operation, and states the maximum temperature limit in the room is 50°C. However, based on applying the Ambient Design Temperature for the DG engine rooms (Tier 1 Section 5 specifies a maximum of 46.1°C) and the DG HVAC Flow Rates (Tier 2 Table 9.4.5.8.2 specifies 160,000 m3/h) as defined in other ABWR DCD sections cited, the DG engine room temperature can exceed this 50°C limit. This departure revises the

DCD Tier 1 Subsection 2.15.5 DG engine room maximum temperature limit during DG operation from 50°C to 60°C.

ABWR DCD Tier 2 Subsections 9.4.5.4.1.2 and 9.4.5.5.5 describe the R/B Safety-Related Electrical Equipment HVAC System and Diesel Generator HVAC System design bases, respectively, including the maximum design temperature limit of the DG Engine rooms. This change also revises Subsections 9.4.5.4.1.2 and 9.4.5.5.5 to state that the indoor temperature in the diesel generator (DG) engine rooms during DG operation is maintained below 60°C.

Evaluation Summary

The proposed change 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, this proposed change consists of increasing the maximum temperature limit in the DG engine rooms during DG operation. It does not change the function or intent of the R/B Safety-Related DG HVAC System or any safety related equipment in the DG engine rooms and therefore does not result in a significant decrease in the level of safety otherwise provided by the design.

(2) This proposed change is consistent with the Atomic Energy Act and other statutes and therefore is authorized by law. As discussed above, this proposed change does not present an undue risk to the public health and safety. This proposed 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 items (ii) and (iv) of 10 CFR 50.12 (a) (2). Specifically, special circumstance (ii) states, “Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.” In this case, the rule is that when the DG is operating, the R/B Safety-Related DG HVAC System and the R/B Safety-Related Electrical Equipment HVAC System maintain the temperature below a specified limit. The DCD Tier 1 Subsection 2.15.5 specifies that the maximum temperature be 50 °C. Because of the Ambient Design Temperature for the DG engine rooms (46.1°C) and the DG HVAC Flow Rates (160,000 m3/h) defined elsewhere in the DCD, the temperature in the DG room can exceed 50°C during DG operation. Therefore, the maximum temperature limit in the DG engine rooms during DG operation requires revision in order to be consistent

with circumstances in the DG engine rooms. Application of the regulation as stated in the Tier 1 Subsection 2.15.5 would therefore not serve the underlying purpose of the rule.

Special circumstance (iv) is also applicable, since this departure changes the design temperature of the DG room to reflect a higher temperature environment. As such, the safety related equipment in this room will be qualified for the higher temperature and therefore will result in a benefit to public health and safety.

(4) This is a "standard" departure that is intended to be applicable to all COL applicants that reference the ABWR DCD. This departure does not adversely affect safety, 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 3.4-1, Safety-Related I&C Architecture

Description

This departure can be characterized into ~~three~~five primary changes.

(1) Elimination of obsolete data communication technology

The departure eliminates references to the Essential Multiplexer System (EMS) and the Non-Essential Multiplexer System (NEMS) originally envisioned in the ABWR architecture and replaces them with separate and independent system level data communication capabilities. The original concept was based on a common EMS, which could be used by multiple safety-related, digitally-based protection systems. This departure defines separate dedicated data communication functions for each safety-related digital platform, including separate and independent data communication functions for each division within a system. The original concerns expressed by the NRC related to the common EMS are addressed as part of Appendix 7A and have been updated to reflect the separate communication capabilities.

In addition, the reference ABWR DCD identified use of the data communication standard ANSI-X3 series, Fiber Distributed Data Interface (FDDI), as the communication protocol for the EMS. FDDI is an obsolete technology and no longer appropriate for use. The safety-related data communication will use a combination of proprietary ~~network~~ data communication protocols and dedicated point-to-point communication to fully meet the defined data communication functional requirements.

The elimination of the multiplexer concept required all references to the system(s) and its primary components to be replaced with a generic data communication reference. The terms EMS and NEMS were eliminated along with Remote Multiplexer Unit (RMU) and Control Room Multiplexer Unit (CMU).

The communication functions are primarily described in FSAR Sections T1 2.2, T1 2.7, T1 3.4, T2 7.2, T2 7.3 and T2 7.9S.

(2) Elimination of unnecessary inadvertent actuation prevention logic and equipment

The reference ABWR DCD described the design of the Engineered Safety Features (ESF) actuation outputs as being fully redundant within each division of the ESF digital controls systems. This design was to minimize the potential for false actuation of ESF components. In the design, each output was processed through two redundant sets of hardware and a final two-out-of-two (2/2) logic decision was to be performed on a component level. Both sets of outputs had to demand actuation before a component would actually respond. As part of the detailed design of the ABWR ESF digital controls, it was determined that only selected ESF components required the redundant actuation prevention logic. If actuated during normal plant operation, most of the ESF components do not have an adverse impact on the safety or operation of the plant. The limited set of components that cannot be actuated during normal operation, such as the main steam isolation valves, are provided with redundant actuation equipment and logic.

The complexity of implementing the fully redundant actuation logic was found to be a detriment to the design, and significantly increased the required maintenance and testing while providing no increase in true plant reliability. As a result, the redundant actuation logic is only implemented for components that may impact plant safety or operation if actuated during normal plant operation.

(3) Clarifications of digital controls nomenclature and systems

The reference ABWR DCD defined many functional design requirements in terms typically reserved for hardware. Examples include the terms “module,” “unit,” and “system.” ~~the terminology was corrected to refer to the requirement as a “function.”~~ The terminology was corrected to refer to the requirement as a “function” to eliminate the confusion associated with purely functional requirements and not physical requirements defined in the DCD. Examples include:

- Digital Trip Module (DTM) to Digital Trip Function (DTF)
- Trip Logic Unit (TLU) to Trip Logic Function (TLF)
- Safety System Logic Unit (SLU) to Safety System Logic Function (SLF)

- Plant Computer System (PCS) to Plant Computer Function (PCF)
- Essential Multiplexer System (EMS) to Essential Communication Function (ECF)

In addition, to better define the functional design and implementation of the digital controls platforms, specific I&C system names were assigned to the ESF digital controls systems and the Reactor Protection System (RPS). The digital controls responsible for the ESF systems are designated as the ESF Logic & Control System (ELCS). The RPS functions are implemented in two separate I&C systems: the Reactor Trip & Isolation System (RTIS) and the Neutron Monitoring System (NMS). The term Safety System Logic & Control (SSLC) was clarified as a general term used to cover all of the logic and controls associated with safety-related control systems.

The nomenclature changes required several sections of the original DCD to be updated for the STP 3&4 COLA to make all sections consistent.

(4) Final selection of platforms changed the implementation architecture

This departure revises the implementation architecture to use configurable logic devices for NMS and RTIS in lieu of microprocessors. These design updates are primarily described in Tier 2 Section 7.2.

(5) Testing and surveillance changes for SSLC

This departure revises the testing and surveillance descriptions for SSLC (NMS, RTIS, ELCS) consistent with the characteristics of the design platforms selected. These changes are primarily described in Tier 2 Section 7.1.

Additionally, the Chapter 16 Technical Specifications Section 3.0 is modified to reflect the above changes to the safety-related I & C architecture.

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) ~~(4)~~ 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) ~~(5)~~ 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) ~~(6)~~ Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since the design change represents another method of accomplishing the underlying purpose of the DCD.

(4) ~~(7)~~ 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 and associated Technical Specification Section 3.0 changes.

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~~specifc data is analyzed to using current methodologies and standards, represents ~~three~~four specific ~~increases~~departures 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~~442.0 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~~1478.3 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.

Wet Bulb 1% Exceedance Values		DCD	STP 3 & 4
	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.

As documented in Sebsections 2.5S.4.4 and 2.5S.4.7, the shear wave velocity at STP 3 & 4 site varies both horizontally in a soil stratum and vertically with elevation, and is lower than the 1,000 ft/sec minimum stated in the DCD. A site specific soil-structure interaction (SSI) analysis will be performed using the measured values of shear wave velocity, with appropriate variation to represent the variability at the site, and site specific SSE, to demonstrate that the results of the site-specific SSI are bounded by the standard plant results included in the DCD. This SSI analysis will be completed and the FSAR will be updated as stated in COM 3A-1.

The liquefaction evaluation documented in Section 2.5S.4.8 uses the measured shear wave velocities, therefore, the results are applicable to STP 3&4 site. Shear wave velocity is not used as an input in the calculation of lateral soil pressures. Therefore, change in shear wave velocity has no impact on calculation of the lateral soil pressures.

The foundation spring constants for mat design are based on settlement calculations. In the development of settlement estimates, the representative shear wave velocity value for intervals within a soil column is only one input used in the derivation of the elastic modulus for layers within that column. Since this derived elastic modulus value is first adjusted for strain and then weighted with estimated values derived from either SPT tests (for granular material) or undrained shear strength tests (for cohesive soils) the effect of variability of shear wave velocity upon settlement calculations is significantly attenuated. Based on this, the foundation spring constants are also relatively insensitive to the variation in shear wave velocity.

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 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. Also, this departure updates Tier 2 to refer to the 2006 International Building Code (IBC), deleting the 1991 Uniform Building Code (UBC). This change incorporates the requirements of Texas building code which adopted 2006 IBC.

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.