entergy

Leia Milster
Manager
Regulatory Assurance
504-739-6250

W3F1-2023-0009

10 CFR 50.71(e)

February 1, 2023

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Technical Specification Index and Bases Update to the NRC for the Period November 11, 2021 through July 25, 2022

Waterford Steam Electric Station, Unit 3
NRC Docket No. 50-382
Renewed Facility Operating License No. NPF-38

In accordance with Waterford Steam Electric Station Unit 3 (Waterford 3) Technical Specification (TS) 6.16, Entergy Operations, Inc. (Entergy) hereby submits an update of all changes made to the Waterford 3 Technical Specification Index and Bases since the last submittal per letter W3F1-2021-0038 (ADAMS Accession No. ML21139A129), dated May 18, 2021. This update should satisfy the submittal frequency required by TS 6.16, which indicates that the submittal will be made at a frequency consistent with 10 CFR 50.71(e). However, during preparation for this submittal, we noted that the requirement was not met. We have entered this into our Corrective Action Program.

Plant changes made under the provisions of 10 CFR 50.59 are reported to the NRC in accordance with the requirements of 10 CFR 50.59(b)(2) by separate submittal.

There are no new commitments associated with this submittal.

Should you have any questions regarding this information, please contact Leia Milster, Manager, Regulatory Assurance, at 504-739-6250.

Respectfully,

Leia Milster

Leia Milster

LEM/llb

Enclosures: 1. Waterford 3 Technical Specification Index and Bases Change List

2. Waterford 3 Technical Specification Index and Bases Revised Pages

cc: NRC Region IV Regional Administrator

NRC Senior Resident Inspector – Waterford Steam Electric Station, Unit 3

NRC Project Manager – Waterford Steam Electric Station, Unit 3

Enclosure 1

W3F1-2023-0009

Technical Specification Index and Bases Change Listing

Waterford 3 Technical Specification Index and Bases Change List

TS Bases Change No.	Implementation Date	Affected TS Index and Bases Pages	Topic of Change
97	11/11/2021	IV VI XIX B 3/4 1-2 B 3/4 1-3 B 3/4 4-1a B 3/4 4-2	Change No. 97 – Nuclear Regulatory Commission (NRC) approved License Amendment 261 relocated the Boration Systems Technical Specifications (TSs) 3.1.2.1 through 3.1.2.8 to the Waterford 3 Technical Requirements Manual (TRM). However, some of the Boration Systems requirements could not be relocated to the TRM because they supported operability of the auxiliary pressurizer spray system. These requirements were retained in the TSs and relocated to Auxiliary Pressurizer Spray TS 3.4.3.2. The TS Index and TS Bases 3/4.1.2 were updated, as appropriate, to reflect the relocation of the Boration Systems TS requirements. (LBDCR 20-009)
98	1/5/2022	B 3/4 7-4a (1)	Change No. 98 – NRC approved License Amendment 218 removed the TS 3.7.6.5 requirement to pressurize the Control Room Envelope (CRE) to 1/8 inch water gauge (w.g.) relative to the outside atmosphere using the Control Room Emergency Actuation Filtration System (CREAFS) in the pressurization mode of operation. This requirement was replaced with the TS 6.5.17 CRE Habitability Program requirement to measure the positive pressure for trending purposes, not as a TS limit. TS Bases 3/4.7.6.1 was updated to remove a reference to the 1/8 inch w.g. limit, which was no longer applicable. (LBDCR 21-048)

TS Bases Change No.	Implementation Date	Affected TS Index and Bases Pages	Topic of Change
99	6/15/2022	XI B 2-2a B 3/4 3-1 B 3/4 3-1a1 B 3/4 3-1c B 3/4 3-1d B 3/4 3-1e B 3/4 3-1f	Change No. 99 – NRC approved License Amendment 260 revised the Waterford 3 TSs to support a modification that replaced the existing digital minicomputers of the Core Protection Calculator System (CPCS) with a digital system based on the Westinghouse Electric Company (Westinghouse) Common Qualified (Common Q) Platform. This modification was needed to address reliability and obsolescence concerns. Multiple TSs were affected by this modification, including the Reactor Trip Setpoints and Reactor Protective Instrumentation TSs, which required changes. TS Bases 2.2.1 and the TS Bases 3/4.3.1 and 3/4.3.2 were revised to update the system requirements reference information and include discussions of the CPCS design features and associated changes to the system operability and testing requirements. The TS Bases Index was revised to incorporate editorial conforming changes. (LBDCR 21-010)
100	6/28/2022	V B 3/4 3-3a B 3/4 3-3b B 3/4 7-4a(4) B 3/4 7-4a(5)	Change No. 100 – NRC approved License Amendment 264 relocated the Chlorine Detection System and the Broad Range Gas Detection system from the TSs to the TRM. The license amendment also supported the relocation of the associated Chemical Detection Systems TS Bases information to the TRM Bases. Accordingly, the TS Index and TS Bases 3/4.3.3.7 were revised to reflect the removal of the Chemical Detection Systems from the TSs and Bases. The CREAFS TS Bases 3/4.7.6.1 were also revised to incorporate conforming changes to reflect the relocation of the Chemical Detection Systems toxic gas signal verification SR. (LBDCR 22-019)

TS Bases Change No.	Implementation Date	Affected TS Index and Bases Pages	Topic of Change
101	7/25/2022	B 3/4 6-7 B 3/4 7-4d	Change No. 101 – NRC approved License Amendment 263 added exceptions to TS SRs when the associated valves and dampers are locked in the actuated position. These TS SR changes were directly related to the information contained in TS Bases 3/4.6.6.1 for the Shield Building Ventilation System and TS Bases 3/4.7.7 for the Controlled Ventilation Area System. Accordingly, these TS Bases sections were revised to incorporate guidance for applying the TS SR exceptions, including the associated operability assessment and restoration requirements. (LBDCR 22-024)

Enclosure 2

W3F1-2023-0009

Technical Specification Index and Bases Revised Pages

(There are 23 unnumbered pages following cover page)

→(DRN 05-747, Ch. 40)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

←(DRN 05-747, Ch. 40)

→(DRN 05-747, Ch. 40)

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

←(DRN 05-747, Ch. 40)

→(DRN 05-747, Ch. 40)

INDEX

LIST OF FIGURES

←(DRN 05-747, Ch. 40)

→LBDCR 20-009, Ch. 97

THIS PAGE IS NOT USED

←LBDCR 20-009, Ch. 97

→LBDCR 20-009, Ch. 97

←LBDCR 20-009, Ch. 97

3/4.1.2.9 BORON DILUTION

This specification is provided to prevent a boron dilution event, and to prevent a loss of SHUTDOWN MARGIN should an inadvertent boron dilution event occur. Due to boron concentration requirements for the RWSP and boric acid makeup tanks, the only possible boron dilution that would remain undetected by the operator occurs from the primary makeup water through the CVCS system. Isolating this potential dilution path or the OPERABILITY of the startup channel high neutron flux alarms, which alert the operator with sufficient time available to take corrective action, ensures that no loss of SHUTDOWN MARGIN and unanticipated criticality occur.

The ACTION requirements specified in the event startup channel high neutron flux alarms are inoperable provide an alternate means to detect boron dilution by monitoring the RCS boron concentration to detect any changes. The frequencies specified in the COLR provide the operator sufficient time to recognize a decrease in boron concentration and take appropriate corrective action without loss of SHUTDOWN MARGIN. More frequent checks are required with more charging pumps in operation due to the higher potential boron dilution rate.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.4.2 SAFETY VALVES

→(LBDCR 20-009, Ch. 97)

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6 x 10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the overpressure protection system provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during reactor shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

←(LBDCR 20-009, Ch. 97)

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized while uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an SIAS test signal the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

The auxiliary pressurizer spray is used to depressurize the RCS by cooling the pressurizer steam space. The auxiliary pressurizer spray is used during those periods when normal pressurizer spray is not available, such as the later stages of a normal RCS cooldown. The auxiliary pressurizer spray also distributes boron to the pressurizer when normal pressurizer spray is not available.

→(LBDCR 20-009, Ch. 97)

The auxiliary pressurizer spray is used, in conjunction with the throttling of the HPSI pumps, during the recovery from a steam generator tube rupture accident. The auxiliary pressurizer spray is also used during a natural circulation cooldown as a safety related means of RCS depressurization to achieve shutdown cooling system initiation conditions and subsequent COLD SHUTDOWN per the requirements of Branch Technical Position (RSB) 5-1. Each train of the auxiliary pressurizer spray system consists of a water supply through either the boric acid makeup pump or the gravity feed valve to the charging pumps and then through the auxiliary pressurizer spray valves. Each train is required to be operable to support the auxiliary pressurizer spray safety function. ACTION "a" is the loss of one train, the remaining train may consist of multiple train components provided auxiliary pressurizer spray flow can still be achieved (e.g., boric acid makeup tank B through the boric acid makeup pumps to charging pump A then discharging through auxiliary pressurizer spray valve B). The loss of both trains or the loss of the water source would require entry into ACTION "b."

←(LBDCR 20-009, Ch. 97)
→(DRN 06-916, Ch. 48)
←(DRN 06-916, Ch. 48)#

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)
→(DRN 04-1223, Ch. 33)

➔(EC-15550, Ch. 59)

3/4.7.6.1 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM (CREAFS) (Continued)

In the pressurization mode, up to 200 cfm outside air is combined with a portion of recirculated air and the combined air flow is filtered, and then added to the air being supplied to the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary. The emergency filtration units are not started in the toxic gas isolation mode.

The normal outside air entering the CRE is continuously monitored by radiation and toxic gas detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state as required. The actions of the toxic gas isolation state are more restrictive and will override the actions of the emergency radiation state.

➔(LBDCR 20-029, Ch. 96; LBDCR 21-048, Ch. 98)

The CREAFS operates at 4225 scfm; all of this can be recirculated air or up to 200 scfm can be outside air. When pressurizing the control room, an emergency outside air path is aligned and dampers are adjusted such that a small portion of the total air being filtered by the CREAFS (up to 200 scfm) is outside air; the remaining air (4025 - 4225 scfm) is from the normal control room HVAC system. After being routed through the emergency filtration unit, the 4225 scfm is returned to the supply duct of the normal control room HVAC system. Up to 200 scfm of outside air is allowed to pressurize the control room. Assuming use of the full 200 scfm of outside air, the air exchange rate would be approximately 6%. The CREAFS operation in maintaining the CRE habitable is discussed in the FSAR, Section 9.4.

← (LBDCR 20-029, Ch. 96; LBDCR 21-048, Ch. 98)

Redundant trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREAFS is designed in accordance with Seismic Category requirements.

➔(LBDCR 20-029, Ch. 96)

The CREAFS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem total effective dose equivalent (TEDE).

← (LBDCR 20-029, Ch. 96)

Applicable Safety Analysis

The CREAFS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access.

The CREAFS provides airborne radiological protection for the CRE as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident fission product release presented in the FSAR, Chapter 15.

← (EC-15550, Ch. 59)

→(DRN 05-747, Ch. 40)

INDEX

BASES

2.2.1 REACTOR TRIP SETPOINTS (Continued)

A Total Loop Uncertainty (TLU) is calculated for each RPS instrument channel. The Trip setpoint is determined by adding or subtracting the TLU from the Analytical Limit (add TLU for decreasing process value; subtract TLU for increasing process value). The Allowable Value is determined by adding an allowance between the Trip Setpoint and the Analytical Limit to account for RPS cabinet Periodic Test Errors (PTE) which are present during a CHANNEL FUNCTIONAL TEST. PTE combines RPS cabinet reference accuracy, calibration equipment errors (M&TE), and RPS cabinet bistable drift. Periodic testing assures that actual setpoints are within their Allowable Values. A channel is inoperable if its actual setpoint is not within its Allowable Value and corrective action must be taken. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the PTE allowance assumed for each trip in the safety analyses.

>(EC-18510, Ch. 64)

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.26 and 21.0 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints. The CPC power adjustment addressable constant BERR1 is used such that the CPC DNBR trip setpoint of 1.26 using the CE-1 critical heat flux correlation assures that the bounding safety limit DNBR of 1.24 for the WSSV-T and ABB-NV correlations will not be exceeded during normal operations and AOOs.

<(EC-18510, Ch. 64)

>(LBDCR 21-010, Ch. 99)

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density -High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in the latest applicable revisions of 00000-ICE-30158, "System Requirements Specification for the Common Q Core Protection Calculator System," Appendix A, as augmented by WNA-DS-04517-CWTR3, "System Requirements Specification for the Core Protection Calculator System, Appendix A."

<(LBDCR 21-010, Ch. 99)

>(EC-26338, Ch. 67)

The Core Protection Calculator, High Logarithmic Power (HLP), and Reactor Coolant System Flow use a single bistable to initiate both the permissive and automatic operating bypass removal functions. A single bistable cannot both energize and de-energize at a single, discrete value due to hysteresis. The CPC automatic bypass removal and permissive for the HLP trip bypass occur at the bistable setpoint (nominally 10^{-4}% power). However, the HLP automatic bypass removal and permissive for CPC trip bypass occur at the reset value of the bistable. Also, note if the bistable setpoint is changed as part of the Special Test Exception 3.10.3, the same dead band transition is applicable.

<(EC-26338, Ch. 67)

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

The OPERABILITY of the Reactor Protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

➔(LBDCR 21-010, Ch. 99)
🡐(LBDCR 21-010, Ch. 99)

➔(LBDCR-14-003 Ch.78)

Table 3.3-1 ACTION 4 requires the suspension of all operations involving positive reactivity changes with the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement. With one of the two required minimum operable channels inoperable, it may not be possible to perform a CHANNEL CHECK to verify the sole remaining required channel is OPERABLE. Therefore, with one or more required channels inoperable, the logarithmic power monitoring function cannot be reliably performed. Consequently, the Required Actions are the same for one required channel inoperable or more than one required channel inoperable.

The (*) for ACTION 4 was added to allow small positive reactivity additions (i.e., temperature or boron fluctuations) necessary to maintain plant conditions. These activities may result in addition to the RCS of water at a temperature different than that of the RCS, may result in slight RCS temperature changes, and may require inventory makeup from sources that are at boron concentrations less than RCS concentration. Depending on core loading and time in core life, raising temperature may add positive reactivity and should be minimized when possible. This allowance is intended to give Operations flexibility to perform actions required to maintain plant conditions but should not be utilized to significantly change plant conditions.

🡐(LBDCR-14-003 Ch.78)

➔(LBDCR 21-010, Ch. 99)

The redundancy design of the Control Element Assembly Calculators (CEACs) in each Core Protection Calculator (CPC) channel (2 CEACs per CPC channel, 8 total CEACs) of the Reactor Protective Instrumentation System maintains CPC channel OPERABILITY as long as one CEAC is OPERABLE in that CPC channel. Table 3.3-1 ACTION 6.c, discussed below, provides actions to maintain a CPC channel OPERABLE with both CEACs inoperable. At least two CPC channels must be OPERABLE to maintain reactor protection. Multiple CEACs may be inoperable in different CPC channels. Actions associated with an inoperable CEAC ensure the affected CPC channel recognizes the condition. Separate actions may be entered for each CPC channel.

🡐(LBDCR 21-010, Ch. 99)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION (Continued)

→(LBDCR 21-010, Ch. 99)

Table 3.3-1 ACTION 6 provides requirements depending on the quantity and combination of inoperable CEACs across the four CPC channels. ACTION 6.a allows for up to two CPC channels to have any one of its two CEAC channels inoperable. The affected CPC channels maintain full functionality as long as the failed CEAC is recognized by the CPC channel via the addressable constant setting.

ACTION 6.b requires additional measures when three or four CPC channels are operating with only a single OPERABLE CEAC in each channel. With three or four CPC channels operating with only a single OPERABLE CEAC, the CEA position verifications ensure the assumptions for using the position values of the target CEAs in each channel remain valid.

ACTION 6.c allows continued operation with both CEACs inoperable in any CPC channel(s) by imposing operational restrictions on CEA position, along with the periodic CEA position verification.

←(LBDCR 21-010, Ch. 99)

→(EC-17731, Ch. 63)

Note 2 of Table 4.3-1 provides requirements for the periodic calibration of CPC power indications using calorimetric power as the calibration standard.

No calibration of CPC power indications are required at less than 15% RATED THERMAL POWER since inherent conservatisms in the CPC calculations at these power levels compensate for any potential decalibration. Significant differences between CPC power indications and calorimetric power observed during surveillances should always be investigated to determine the cause of the deviation. The most accurate calorimetric power indication available at the time of calibration should be used.

Between 15% and 80% power, if the daily surveillance finds that a CPC power indication is greater than the calorimetric power indication by more than 10% RTP, it should be adjusted to be within 8.0% and 10.0% RTP above the calorimetric. If the CPC power indications have been calibrated properly to the calorimetric power indication at high power (meaning 80% or above), then the most appropriate thing to do is not calibrate CPC powers below 80% power if they are conservative relative to calorimetric. In the extremely unlikely event that a CPC power indication is found to be more than 10.0% RTP higher than the calorimetric, it should be adjusted as little as possible to meet the requirements of the Technical Specifications. If this situation were to occur, it is likely that there is an anomaly in the calibration data or instrumentation. The safety and setpoint analysis does not explicitly address this situation because it is an unreasonable scenario without some other anomaly in the measurements, calibration or instrumentation. The

←(EC-17731, Ch. 63)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION (Continued)

When one of the inoperable channels is restored to OPERABLE status, subsequent operation in the applicable MODE(S) may continue in accordance with the provisions of ACTION 19.

Because of the interaction between process measurement circuits and associated functional units as listed in the ACTIONS 19 and 20, placement of an inoperable channel of Steam Generator Level in the bypass or trip condition results in corresponding placements of Steam Generator ΔP (EFAS) instrumentation. Depending on the number of applicable inoperable channels, the provisions of ACTIONS 19 and 20 and the aforesaid scenarios for Steam Generator ΔP (EFAS) would govern.

➔(LBDCR 16-046, Ch. 86)

The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the frequencies in the Surveillance Frequency Control Program are sufficient to demonstrate this capability. The frequency for the channel functional tests for these systems is controlled by the Surveillance Frequency Control Program.

➔(LBDCR 16-046, Ch. 86)

➔(LBDCR 21-010, Ch. 99)

The CPC testing features are designed to allow for complete testing by using a combination of system self-checking and manual tests. Successful testing consists of verifying that the capability of the system to perform the safety function has not failed or degraded. For hardware functions this would involve verifying that the hardware components and connections have not failed or degraded. Software testing involves verifying that the software code has not changed and that the software code is executing. To the extent possible, CPC system testing will be accomplished with continuous system self-checking features in lieu of manual surveillance tests. Self-checking features include on-line diagnostics for the computer system and the hardware and communications tests. Faults detected by the self-checking features are alarmed in the main control room. These self-checking tests do not interfere with normal system operation. The performance of channel checks validates that the self-diagnostics are continuing to perform their self-checking functions.

←(LBDCR 21-010, Ch. 99)

➔(LBDCR 16-046, Ch. 86)

Testing frequency for the Reactor Trip Breakers (RTBs) is controlled by the Surveillance Frequency Control Program. The RTB channel functional test and RPS logic channel functional test are scheduled and performed such that RTBs are verified OPERABLE in accordance with the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

RPS/ESFAS Trip Setpoints values are determined by means of an explicit setpoint calculation analysis. A Total Loop Uncertainty (TLU) is calculated for each RPS/ESFAS instrument channel. The Trip Setpoint is then determined by adding or subtracting the TLU from the Analytical Limit (add TLU for decreasing process value; subtract TLU for increasing process value). The Allowable Value is determined by adding an allowance between the Trip Setpoint and the Analytical Limit to account for RPS/ESFAS cabinet Periodic Test Errors (PTE) which are present during a CHANNEL FUNCTIONAL TEST. PTE combines the RPS/ESFAS

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION (Continued)

cabinet reference accuracy, calibration equipment errors (M&TE), and RPS/ESFAS cabinet bistable Drift. Periodic testing assures that actual setpoints are within their Allowable Values. A channel is inoperable if its actual setpoint is not within its Allowable Value and corrective action must be taken. Operation with a trip set less conservative than its setpoint, but within its specified ALLOWABLE VALUE is acceptable on the basis that the difference between each trip Setpoint and the ALLOWABLE VALUE is equal to or less than the Periodic Test Error allowance assumed for each trip in the safety analyses.

→(EC-26338, Ch. 67)

The Core Protection Calculator, High Logarithmic Power (HLP), and Reactor Coolant System Flow use a single bistable to initiate both the permissive and automatic operating bypass removal functions. A single bistable cannot both energize and de-energize at a single, discrete value due to hysteresis. The CPC automatic bypass removal and permissive for the HLP trip bypass occur at the bistable setpoint (nominally 10^{-4}% power). However, the HLP automatic bypass removal and permissive for CPC trip bypass occur at the reset value of the bistable. Also note if the bistable setpoint is changed as part of the Special Test Exception 3.10.3, the same dead band transition is applicable.

←(EC-26338, Ch. 67)

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be verified by any series of sequential, overlapping, or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Topical Report CE NPSD-1167-A, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the topical report. Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.

→(LBDCR 21-010, Ch. 99)

WCAP-18484-P, "Licensing Technical Report for the Waterford Steam Electric Station Unit 3 Common Q Core Protection Calculator System," Appendix B, "Elimination of Specific CPCS Technical Specification Surveillance Requirements," provides the basis and methodology for using allocated CPCS digital equipment response times in the overall verification of the channel response time for the CPCS. Response time verification for other equipment within the CPCS channel must be demonstrated by test as identified in the Technical Specifications.

←(LBDCR 21-010, Ch. 99)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION (Continued)

→(EC-26338, Ch. 67)

In the applicable logarithmic power modes, with the Logarithmic Power circuit inoperable or in test, the associated functional units of Local Power Density-High, DNBR-Low, and Reactor Coolant Flow-Low should be placed in the bypassed or tripped condition. With logarithmic power greater than 10^{-4}% bistable setpoint and Local Power Density-High, DNBR-Low, and Reactor Coolant Flow-Low no longer bypassed (either through automatic or manual action), these functional units may be considered OPERABLE.

← (EC-26338, Ch. 67)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

TABLE 3.3-1, Functional Unit 13, Reactor Trip Breakers

The Reactor Trip Breakers Functional Unit in Table 3.3-1 refers to the reactor trip breaker channels. There are four reactor trip breaker channels. Two reactor trip breaker channels with a coincident trip logic of one-out-of-two taken twice (reactor trip breaker channels A or B, and C or D) are required to produce a trip. Each reactor trip breaker channel consists of two reactor trip breakers. For a reactor trip breaker channel to be considered OPERABLE, both of the reactor trip breakers of that reactor trip breaker channel must be capable of performing their safety function (disrupting the flow of power in its respective trip leg). The safety function is satisfied when the reactor trip breaker is capable of automatically opening, or otherwise opened or racked-out.

If a racked-in reactor trip breaker is not capable of automatically opening, the ACTION for an inoperable reactor trip breaker channel shall be entered. The ACTION shall not be exited unless the reactor trip breaker capability to automatically open is restored, or the reactor trip breaker is opened or racked-out.

→(EC-12084, Ch. 57)

TABLES 3.3-3 and 4.3-2, Functional Unit 6, Loss of Power (LOV)

The Loss of Power Functional Unit 6 in Tables 3.3-3 and 4.3-2 refers to the undervoltage relay channels that detect a loss of bus voltage on the 4kV (A3 & B3) and 480V (A31 & B31) safety buses and a sustained degraded voltage condition on 4kV (A3 & B3) safety buses. The intent of these relays is to ensure that the Emergency Diesel Generator starts on a loss of voltage or a sustained degraded voltage condition. The response time SR in TS 3.3.2 ensures that Bus A3 and B3 undervoltage relays trip and generate a Loss of Voltage (LOV) signal in 2 seconds for initiation of the EDG start. The response time for Bus AB3 and AB31 relays is not as critical as the Bus A3 and B3 undervoltage relays. Bus AB3 and AB31 undervoltage relays [4KVEREL3AB-1A(1B)(1C) and SSDEREL31AB-1A(1B)(1C)] strip bus loads upon an undervoltage condition to preclude any perturbations which might affect the A and B buses and prepare the bus to be energized by an EDG with subsequent loading by the sequencer. Bus AB3 and AB31 undervoltage relays do not provide an EDG start signal. Therefore, TS 3/4.3.2, Tables 3.3-3 and 4.3-2 Functional Unit 6 requirements, are not applicable to AB3 Bus and AB31 Bus undervoltage relays.

If an AB Bus undervoltage relay becomes inoperable, initiate a condition report and consider operability of the associated EDG based on the AB Bus loads when evaluating the failure.

← (EC-12084, Ch. 57)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

(DRN 05-747, Ch. 40)

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

(DRN 05-747, Ch. 40)

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

Continuous operation with less than Minimum Channels OPERABLE requirements is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the accident monitoring instrumentation. Therefore, requiring restoration of one inoperable channel limits the risk that the variable will be in a degraded condition should an accident occur. If the 7 day requirement is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 4 within 12 hours. The completion time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

TS 3/4.3.3.6 applies to the following instrumentation: ESFIPI6750 A, ESFIPR6750 B, ESFIPR6755 A&B, RC ITI0122 HA, RC ITI0112 HB, RC ITI0122 CA, RC ITI0112 CB, RC IPI0102 A,B,C,&D, RC ILI0110 X&Y, SG ILI1113 A,B,C,&D, SG ILI1123 A,B,C,&D, SG ILI1115 A2&B2, SG ILI1125 A2&B2, SI ILI7145 A, SI ILR7145 B, all CET's, all Category 1 Containment Isolation Valve Position Indicators, EFWILI9013 A&B, HJTC's, and ENIIJI0001 C&D.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→(LBDCR 22-019, Ch. 100)

3/4.3.3.7 This section deleted

←(LBDCR 22-019, Ch. 100)

→(LBDCR 22-019, Ch. 100)
←(LBDCR 22-019, Ch. 100)

3/4.3.3.8 This section deleted

3/4.3.3.9 This section deleted

→(EC-15550, Ch. 59)

3/4.7.6.1 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM (CREAFS) (Continued)

reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day completion time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day completion time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

ACTION STATEMENT c requires that, in MODE 1, 2, 3, or 4, if the inoperable CREAFS or the CRE boundary cannot be restored to OPERABLE status within the required completion time, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

ACTION STATEMENT d.1 requires that, in MODE 5 or 6, or during movement of irradiated fuel assemblies, if required Action a cannot be completed within the required completion time, the OPERABLE CREAFS train must be immediately placed in the emergency radiation protection mode of operation. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

ACTION STATEMENT d.2 is an alternative to Action d.1 and requires immediate suspension of activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position.

→(LBDCR 20-029, Ch. 96) (LBDCR 22-019, Ch. 100)

ACTION STATEMENT e requires that, in MODES 5 or 6, or during movement of irradiated fuel assemblies, with one or more CREAFS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position. SR 4.7.6.1.d.2 includes an exception that allows the SR to be considered met for dampers and valves that are locked, sealed, or otherwise secured in the actuated position. As a result, LCO 3.7.6.1 Action e is not required to be entered in this situation, and the CRE boundary remains operable. The exception may only be applied to dampers and valves that have demonstrated acceptable leakage performance.

←(LBDCR 20-029, Ch. 96) (LBDCR 22-019, Ch. 100)

ACTION STATEMENT f addresses the condition of both CREAFS trains being inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary. The CREAFS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

ACTION STATEMENT g requires that, in MODES 5 or 6, or during movement of irradiated fuel assemblies, with both CREAFS trains inoperable action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

←(EC-15550, Ch. 59)

➔(EC-15550, Ch. 59)

3/4.7.6.1 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM (CREAFS) (Continued)

Surveillance Requirements

➔(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

🡐(LBDCR 16-046, Ch. 86)

➔(LBDCR 20-029, Ch. 96) (LBDCR 22-019, Ch. 100)

d. SR 4.7.6.1.d.2 includes an exception that allows the SR to be considered met for dampers and valves that are locked, sealed, or otherwise secured in the actuated position in Modes 5, 6, or defueled. As a result, LCO 3.7.6.1 Action e. is not required to be entered in this situation, and the CRE boundary remains operable. The exception may only be applied to dampers and valves that have demonstrated acceptable leakage performance because operability of the CRE boundary is contingent on the leak-tightness of the CRE boundary dampers and valves. Acceptable leakage performance is demonstrated by satisfactory performance of the control room tracer gas test, differential pressure testing, component testing for HVCMVAAA101 and HVCMVAAA102, and completion of scheduled preventative maintenance tasks. It is understood that component testing of the remaining twelve boundary valves is not possible due to system configuration; thus, individual component testing beyond the unfiltered inleakage and differential pressure testing is not required to be performed on these valves to demonstrate leak-tightness.

🡐(LBDCR 20-029, Ch. 96) (LBDCR 22-019, Ch. 100)

g. This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is not greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Action b must be entered. Action b.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3 (Ref. 1) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 2). These compensatory measures may also be used as mitigating actions as required by Action b.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 3). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

References

1. Regulatory Guide 1.196
2. NEI 99-03, "Control Room Habitability Assessment," June 2001
3. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI), dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability," (ADAMS Accession No. ML040300694)

3/4.7.6.2 [NOT USED]

🡐(EC-15550, Ch. 59)

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SHIELD BUILDING VENTILATION SYSTEM

→ (DRN 05-131, Ch. 39)

The OPERABILITY of the shield building ventilation systems ensures that containment vessel leakage occurring during design basis accidents into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 50.67.

← (DRN 05-131, Ch. 39)

Acceptable removal efficiency is shown by a methyl iodide penetration of less than 0.5% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 70%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analyses and is consistent with Regulatory Guide 1.52 and ASTM D3803-1989.

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

→(LBDCR 22-024, Ch. 101)

SRs 4.6.6.1.d.2 and d.3 exclude automatic dampers and/or valves that are locked, sealed, or otherwise secured in the actuated position. The SRs do not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the operability of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency.

←(LBDCR 22-024, Ch. 101)

3/4.6.6.2 SHIELD BUILDING INTEGRITY

→ (DRN 05-131, Ch. 39)

SHIELD BUILDING INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the shield building ventilation system, will limit the site boundary radiation doses to within the limits of 10 CFR 50.67 during accident conditions.

← (DRN 05-131, Ch. 39)

→(LBDCR 16-046, Ch. 86)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

←(LBDCR 16-046, Ch. 86)

3/4.7.7 CONTROLLED VENTILATION AREA SYSTEM (Continued)

Acceptable removal efficiency is shown by a methyl iodide penetration of less than 0.5% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 70%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

➔(LBDCR 16-046, Ch. 86)

Operation of the system with the heaters on for at least 10 hours continuous in accordance with the Surveillance Frequency Control Program is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analyses and is consistent with Regulatory Guide 1.52 and ASTM D3803-1989.

The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.

🡐(LBDCR 16-046, Ch. 86)

➔(LBDCR 22-024, Ch. 101)

SRs 4.7.7.d.2 and d.3 exclude automatic dampers and/or valves that are locked, sealed, or otherwise secured in the actuated position. The SRs do not apply to dampers or valves that are locked, sealed, or otherwise secured in the actuated position since the affected dampers or valves were verified to be in the actuated position prior to being locked, sealed, or otherwise secured. Placing an automatic valve or damper in a locked, sealed, or otherwise secured position requires an assessment of the operability of the system or any supported systems, including whether it is necessary for the valve or damper to be repositioned to the non-actuated position to support the accident analysis. Restoration of an automatic valve or damper to the non-actuated position requires verification that the SR has been met within its required Frequency.

🡐(LBDCR 22-046, Ch. 101)