Entergy

Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213
Tel 601-368-5138

Ron Gaston
Director, Nuclear Licensing

10 CFR 50.90

W3F1-2021-0032

May 21, 2021

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

Subject: Revised License Amendment Request - Digital Upgrade to the Core Protection Calculator (CPC) System and Control Element Assembly Calculator (CEAC) System

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

References:

1. Entergy Operations, Inc. (Entergy) letter to U.S. Nuclear Regulatory Commission (NRC), "License Amendment Request to Implement a Digital Upgrade to the Core Protection Calculator (CPC) system and Control Element Assembly Calculator (CEAC) system," dated July 23, 2020, (ADAMS Accession No. ML20205L587)

2. Entergy letter to NRC, "Revised Vendor Oversight Plan Summary - License Amendment Request to Implement a Digital Upgrade to the Core Protection Calculator (CPC) System and Control Element Assembly Calculator (CEAC) System," dated January 29, 2021, (ADAMS Accession No. ML21029A156)

In Reference 1, Entergy Operations, Inc. (Entergy) submitted, to the U.S. Nuclear Regulatory Commission (NRC), a proposed amendment to Appendix A, "Technical Specifications" (TS) of Renewed Facility Operating License No. NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford). The proposed change would revise the Waterford TS in order to implement a planned digital instrumentation and control (DI&C) modification of the Core Protection Calculator (CPC) system and Control Element Assembly Calculator (CEAC) system.

Based on ongoing discussions with the NRC concerning the Reference 1 license amendment request (LAR), Entergy has revised the Enclosure to the LAR (i.e., the "Evaluation of the Proposed Change"), including Attachment 1, "Technical Specification Page Markups," and Attachment 2, "Clean Technical Specification Pages." The Enclosure to this letter provides this revision. Attachments 1 and 2 in this submittal replace, in their entirety, Attachments 1 and 2 the original Reference 1 LAR. Attachments 3 through 13 and Attachment 15 of the original Reference 1 LAR remain applicable.

In Reference 2, Entergy transmitted Revision 1 of the Entergy Vendor Oversight Plan (VOP) Summary for the CPC/CEAC replacement project. This was provided as a replacement to Attachment 14 of the Reference 1 LAR.

Changes to the original "Evaluation of the Proposed Change" are noted with underlined text and revision bars in the right-hand margin.

The No Significant Hazards Consideration determination provided in the Referenced LAR submittal is not altered by the information provided in this letter.

There are no new regulatory commitments included in this letter.

In accordance with 10 CFR 50.91(b)(1), "Notice for public comment; State consultation," a copy of this letter, without the proprietary attachments, is being provided to the designated State Official.

Should you have any questions or require additional information, please contact Paul Wood, Regulatory Assurance Manager, Waterford, at (504) 464-3786 or pwood1@entergy.com.

I declare under penalty of perjury, that the foregoing is true and correct. Executed on May 21, 2021.

Respectfully,

Ron W. Gaston

Ron Gaston

RWG/jls

Enclosure: Evaluation of the Proposed Change, Revision 1
Attachment 1: Technical Specification Page Markups, Revision 1
Attachment 2: Clean Technical Specification Pages, Revision 1

cc: NRC Region IV Regional Administrator
NRC Senior Resident Inspector - Waterford Steam Electric Station, Unit 3
Louisiana Department of Environmental Quality
NRC Project Manager - Waterford Steam Electric Station, Unit 3

Enclosure

W3F1-2021-0032

Evaluation of the Proposed Change
Revision 1

Evaluation of the Proposed Change
Revision 1

1. SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, Entergy Operations, Inc. (Entergy) requests an amendment to Appendix A, "Technical Specifications" (TS) of Renewed Facility Operating License No. NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford). The proposed change will revise the Waterford TS in order to implement a planned digital modification at Waterford.
The following TS sections are affected by this change:

- TS 2.2.1 Reactor Trip Setpoints
- TS 3.1.3 CEA Position
- TS 3.2.4 DNBR Margin
- TS 3.3.1 Reactor Protective Instrumentation
- TS 3.10.2 Moderator Temperature Coefficient, Group Height, Insertion, and Power Distribution Limits
- TS 6.8.1 Procedures and Programs
- TS 6.9 Reporting Requirements

The modification will replace the existing digital minicomputers of the Core Protection Calculator (CPC) system and Control Element Assembly Calculator (CEAC) system with a more reliable, digital system based on the Westinghouse Electric Company (Westinghouse) Common Qualified (Common Q) Platform. The Core Protection Calculator System (CPCS) is the combined CPC and CEAC. The Common Q platform has an NRC-approved topical report (Reference 11).

Waterford is the only nuclear site utilizing the original version of the CPCS. An Interdata 7/16 computer system is used in four channels of the CPCS. There are obsolescence concerns with the equipment due to limited spare parts availability. In addition, there are reliability concerns due to the identification of single point vulnerabilities in the system.

In Reference 1, Entergy submitted a letter-of-intent (LOI) to the U.S. Nuclear Regulatory Commission (NRC) that described a planned DI&IC license amendment request (LAR) for the CPCS modification at Waterford, indicating that the LAR would be developed and submitted in accordance with the Alternate Review Process (ARP) guidance in NRC DI&C Interim Staff Guidance (ISG)-06, "Licensing Process," Revision 2 (Reference 2). The LAR format and contents are consistent with the DI&C-ISG-06 guidance for the ARP.

Entergy plans to implement the digital upgrade modification to the CPC and CEAC systems at Waterford during the 24th refueling outage (RF24), which is scheduled for Spring 2022. In order to initiate and complete equipment fabrication and factory acceptance testing prior to the start of the refueling outage, Entergy requests approval of the proposed license amendment by August 24, 2021. The proposed changes will be implemented prior to start-up from RF24.

2. DETAILED DESCRIPTION

1. System Design and Operation

The Waterford Plant Protection System (PPS) is comprised of an Engineered Safety Features Actuation System (ESFAS) and a Reactor Protection System (RPS). The Core Protection Calculator System (CPCS) is part of the RPS.

The CPC/CEAC system issues two reactor trip signals to the RPS to protect the fuel design limits. These four independent Core Protection Calculators (CPCs), one in each protection channel, calculates departure from nucleate boiling ratio (DNBR) and local power density (LPD). The reactor trips provided by the CPCs are inputs to the RPS Coincidence and Initiation Logic. The CPC trips have a 2 out of 4 logic.

The calculations are performed in each CPC, utilizing the following input signals:

- Core inlet and outlet temperature,
- Pressurizer pressure,
- Reactor coolant pump speed,
- Excore nuclear instrumentation flux power (each subchannel from the safety channel),
- Selected (target) CEA position, and
- CEA subgroup deviation from the CEA calculators.

The DNBR and LPD calculation results are compared to trip setpoints for initiation of a low DNBR trip and a high LPD trip. These CPCS trip outputs become digital trip inputs to the corresponding RPS channel. The four channel RPS performs the 2 out of 4 coincidence logic on various reactor trip functions that include the CPC Low DNBR and High LPD. The CPCS is designed to initiate automatic protective action to assure that the specified acceptable fuel design limits (SAFDL) on DNBR and LPD are not exceeded during Anticipated Operational Occurrences (AOOs).

The High LPD Trip is to prevent the linear heat rate (kW/ft) in the limiting fuel pin in the core from exceeding the value corresponding to the centerline fuel melting temperature. This is to prevent exceeding the safety limit of peak fuel centerline temperature in the event of defined anticipated operational occurrences.

DNBR is the ratio of Critical Heat Flux to Actual Heat Flux. Critical heat flux (CHF) is that value of heat flux at which Departure from Nucleate Boiling (DNB) occurs. The Low DNBR trip is to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit for the fuel cladding in the event of defined anticipated operational occurrences. In addition, this trip will provide a reactor trip to assist the Engineered Safety Features System (ESFS) in limiting the consequences of the steam generator tube rupture, steam line break and reactor coolant pump shaft seizure accidents.

CPC DNBR and LPD pre-trip alarms are initiated prior to the trip value to provide audible and visible indication of approach to a trip condition. These pre-trip functions have no direct safety function.

The CPC will also initiate DNBR and LPD trip outputs (i.e., Auxiliary trips) under the following conditions:

- CPC operating space limits are exceeded for the hot pin axial shape index, integrated one pin radial peak, maximum and minimum cold leg temperatures, and primary pressure (CPC Operating Space Trips).
- Opposing cold leg temperature difference exceeds its setpoint, which varies with power level (Asymmetrical Steam Generator Transient (ASGT) Trip).
- Reactor power exceeds the variable overpower trip setpoint. The trip setpoint is larger than the steady state reactor power by a constant offset. However, it is limited in how fast it can follow changes in reactor power. This provides protection from sudden power increases (Variable Overpower Trip)
- The maximum hot leg temperature approaches the coolant saturation temperature (Thot at saturation).
- The CPC system is not set in the normal operating configuration (CPC Failure).
- Reactor coolant pump shaft speed drops below its setpoint value for multiple pumps (Less than two RCPs running).

The CPCS/CEAC design basis functions are not changing as a result of this CPCS modification. All the design basis events in Chapter 15 and the reliance on the CPCS low DNBR and high LPD trips are unchanged.

The PPS/RPS performs a two out of four coincidence of like trip signals to generate a reactor trip signal. The use of four channels allows bypassing of one channel for maintenance while maintaining a two out of three channel trip.

The scope of this modification is the replacement of the CPCS including sensor terminations, replacement calculators (CPC and CEAC), alarm output termination, analog output terminations (Main Control Room (MCR) Indication), and output terminations to the PPS/RPS.

Excluded from the CPCS modification are:

- Sensors and their cabling to the CPCs
- Reactor Protection System
- CPC system Trip setpoints and outputs.

All functional requirements for DNBR and LPD trip output are unchanged.

2. Current TS Requirements

The following Technical Specifications (TS) sections are affected by this change:

2.2.1	Reactor Trip Setpoints
3/4.1.3.1	CEA Position
3.2.4	DNBR Margin
3/4.3.1	Reactor Protective Instrumentation
3/4.10.2	Moderator Temperature Coefficient, Group Height, Insertion, and Power Distribution Limits
6.8.1	Procedures and Programs
6.9	Reporting Requirements

TS 2.2.1 provides the list of reactor protective instrumentation setpoints in Table 2.2-1. None of the CPC-related setpoints are affected by the proposed changes, as discussed in section 2.4 below.

TS 3.1.3.1 provides the operability and alignment requirements for the Core Element Assemblies (CEAs) groups. Surveillance Requirement 4.1.3.1.1 specifies when the alignment checks are performed depending on CEAC operability status.

TS 3.2.4 provides requirements for monitoring DNBR Margin depending on the status of Core Operating Limits Supervisory System (COLSS) and CEACs.

TS 3.3.1 provides minimum operability requirements for the reactor protective instrumentation which includes CPCs and CEACs.

TS 3.10.2 provides the requirements for a special test exception permitting individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of select physics tests.

TS 6.8.1 is an administrative TS that governs modifications to CPCS software.

TS 6.9 is an administrative TS that governs reporting requirements.

3. Reason for the Proposed TS Changes

There are three aspects of the CPCS modification that drive the proposed changes:

2 to 8 CEAC Design Change

Many of the changes are due to the configuration change from having two CEACs shared across the four CPC channels to two dedicated CEACs in each of the four CPC channels. Some of the necessary changes are editorial, since currently the term "BOTH CEACs" applies to all CEAC capability and in the new configuration it does not. Having eight total CEACs also greatly reduces the operational impact of individual CEACs being inoperable.

Common Q Design

Due to the Common Q design, CPC features that are currently part of the Waterford TS are no longer applicable. For example, Surveillance Requirement (SR) 4.3.1.5 contains requirements for determining CPC or CEAC operability following three auto restarts. The upgraded CPCs will not have an auto restart function, thereby rendering this SR obsolete and no longer applicable.

Crediting Self-Diagnostics for TS Surveillance Requirement Elimination

The Common Q design also provides additional reliability and operational margin via the self-diagnostics. These self-diagnostics are continually monitoring the health of the hardware and software. Appendix B to the Licensing Technical Report (LTR) (Attachment 4) and the Waterford System Engineer and Operations actions supporting TS SR reduction (i.e., as described in Section 3.4 below) provide the justification to remove selected SRs.

4. Description of the Proposed TS Changes

Changes are proposed to the following Technical Specifications (TS) as described in the table below. TS markups are provided in Attachment 1.

2.2.1	Reactor Trip Setpoints
3.1.3.1	CEA Position
3.2.4	DNBR Margin
3.3.1	Reactor Protective Instrumentation
3.10.2	Moderator Temperature Coefficient, Group Height, Insertion, and Power Distribution Limits
6.8.1	Procedures and Programs
6.9	Reporting Requirements

TS Section	Proposed Change
TS 2.2.1 Table 2.1	The proposed changes to TS 2.2.1 are confined to Table 2.2-1. The changes are predominantly editorial to conform to the updated CPC-to-CEAC relationship, where two CEACs are provided in each CPC channel. The CPCs are the primary functional unit, possessing two trip functions, LPD-High and DNBR- Low. The culmination of the change is that the CPCs are Functional Unit 9, with the two trips listed. The former functional units 10, 14 and 15 are marked as "DELETED". Since the CEACs provide no direct trip function, they are not listed in the revised Table 2.2-1. However, since CEACs have operability and surveillance requirements they are included in Tables 3.3-1 and 4.3.1. None of the CPC-related setpoints are affected by the proposed changes.

TS 3.1.3.1 SR 4.1.3.1.1	The Surveillance Requirement 4.1.3.1.1 listed in TS 3.1.3.1 contains the only change to this TS. The operability requirements of the CEAs are not impacted. The objective of the SR is also unchanged. The proposed change removes the current TS guidance on how often the SR should be performed depending on the operability condition of the CEACs. This guidance is redundant to the proposed TS 3.3.1 Action 6 statement which dictates when CEA position checks are performed depending on CEAC operability status. As described below, Action 6 directly stipulates performance of SR 4.1.3.1.1 on the same 4 hour frequency as is currently required.
TS 3.2.4	TS 3.2.4 is reformatted to resemble the PVNGS TS 3.2.4 wording, by grouping the four methods of monitoring DNBR depending on the status of the Core Operating Limit Supervisory System (COLSS). The PVNGS LCO wording was chosen because it concisely handles the eight CEAC configuration design and functionality impacts. It was previously reviewed and approved by the NRC, which is described in Section 4.2, "Precedent". The actions to take when the DNBR limit is not maintained are unchanged from the present Waterford TS 3.2.4.
TS 3.3.1 Table 3.3-1 including Table Notation	The Functional Unit designations are changed, similarly to Table 2.2-1 to put all the CPC subfunctions under Functional Unit 9, Core Protection Calculators (LPD – High, DNBR – Low and CEACs). The table requirements for the CPC, LPD, and DNBR are identical, and are listed as a single line entry. Notation "(h)" was added under the "Channels to Trip" column. The CEACs are included under Functional Unit 9 because each pair of CEACs directly supports one of the four CPC channels. Also, the "Total No. of Channels", "Channels to Trip", "Minimum Channels OPERABLE", and "Action" values were changed to reflect the eight CEAC configuration: • Total No. of Channels – In the new CPC design, each of the four CPC channels houses a dedicated pair of CEACs. Therefore, there are four channels of CEACs, with two CEACs per Channel. Reference to notations "(g)" and "(i)" are also added.

	• Channels to Trip – CEACs cause trips by transmitting a high penalty factor (PF) to its associated CPC channel. It requires two CPC channels to trip on either LPD – High or DNBR – Low to cause a reactor trip. Therefore, two separate channels of CEACs must send sufficiently high penalty Factor (PF) to their CPC to cause a reactor trip. • Minimum Channels Operable – A channel of CEAC is OPERABLE as long as one of the two CEACs in a CPC channel are OPERABLE. Therefore, requiring three channels as a minimum to be OPERABLE matches the CPC requirements and ensures single failure criteria is maintained or ACTIONS taken. Reference to notations "(g)" and "(i)" are also added. Table 3.3-1, Table Notation, notes (g), (h), and (i) were added. These provide clarifying information concerning CEAC and CPC operability: • (g) There are two CEACS in each CPC channel. • (h) Both Local Power Density – High and DNBR – Low must be OPERABLE for a CPC Channel to be OPERABLE. • Both CEACs in an inoperable CPC channel are also inoperable.
TS 3.3.1 Table 3.3-1 Action Statements	<u>Action 6</u> Action 6 is revised to accommodate the eight CEAC configuration, while maintaining essentially the same actions as the current TS, depending on the impact to CPCS functionality. A primary objective of the proposed changes to Action 6 is to ensure that all CEAC conditions of operability are included. For all of the actions described below, there is the option of declaring the associated CPC channel inoperable, which would invoke Actions 2 or 3, which are unchanged. The current Action 6 only contains two parts (one CEAC inoperable and both CEACs inoperable). In the proposed changes, considering the eight CEAC design, there are multiple combinations of potential CEAC inoperability, with varying impacts to CPCS functionality. To utilize the operational flexibility and redundancy offered by eight

	CEACs, while maintaining an understandable presentation of the Actions, the current two part Action 6 is being revised to describe three CEAC operability conditions. The addition of a NOTE indicates that separate entries may be made for each CPC. Action "a" is new and reflects the robustness of the CPCS design such that up to two CPC channels maintain full capability with only a single CEAC OPERABLE in each. The action consists of ensuring the affected CPC channels does not use the input from the failed CEAC by manually setting the appropriate addressable constant. From a safety function perspective, the CPCS is fully capable of meeting all functional requirements. This is because the CEAs in each subgroup are monitored by redundant reed switch position transmitters (RSPT 1 and RSPT 2). CEAC 1 in each CPCS channel is identical and therefore redundant in four CPCS channels. It monitors all the CEA RSPT 1 signals to compute a penalty factor for the CPC in case there is a CEA deviation in a subgroup. Similarly, CEAC 2 in each CPCS channel is identical and therefore redundant in four CPCS channels. It monitors all CEA RSPT 2 signals to compute a penalty factor for the CPC in case there is a CEA deviation in a subgroup. If CEAC 1 or CEAC 2 is inoperable in a CPCS channel, the operable CEAC can still compute a CEA deviation penalty factor for the CPC using either RSPT 1 or RSPT 2 signals depending on the CEAC that is still operable in the channel. If two CPCS channels have 1 CEAC inoperable, the worst case scenario is that the same CEAC is inoperable in both CPCS channels. For example, if CEAC 1 is inoperable in both CPCS Channel A and Channel B, then the CPC in those channels rely solely on CEAC 2 to compute the CEA deviation penalty factor based on RSPT 2 signals. If we postulate an undetected error in one of the CEAC 2's in Channel A or B, as required by IEEE 603, Clause 5.1, the 4-channel CPCS is still able to perform its safety function because it has 2 channels that have 2 operable CEACs (Channels C and D), and 1 channel with 1 operable CEAC. These three channels can calculate a CEA deviation penalty factor for the CPC. Should failures in the RSPTs occur that causes a CEAC to fail, this failure would cause CEAC failures to occur in all four CPCS channels which exceeds the condition of two CPCS channels having 1 CEAC inoperable. In the

	case of an undetected RSPT failure (e.g., RSPT1), this scenario affects 1 CEAC in all four CPCS channels. The other CEAC (e.g., CEAC 2) can still perform its safety function by generating a penalty factor based on the redundant RSPT signal (e.g., RSPT2). Action "b" is similar to the current TS action 6 for a single CEAC inoperable. It provides additional requirements when the third or fourth CPC channel experiences the inoperability of one of the two contained CEACs. Action "b.1" ensures the CPC channel does not use the input from the failed CEAC by setting the appropriate addressable constant. Action "b.2" is similar to the current action "6a" except instead of describing the 4-hour action similar to SR 4.1.3.1.1, it directs the performance of that SR. Action "c" is similar to the current set of "6c" actions, including specifying the 4-hour CEA position checks via performance of SR 4.1.3.1.1. <u>Action 7</u> Action 7 is being deleted since it is associated with Auto-restarts of the CEAC which is not a function of the upgraded system.
TS 3.3.1 SR 4.3.1.3	SR 4.3.1.3 is modified to also exclude CPC and CEAC, along with neutron detectors, from REACTOR TRIP SYSTEM RESPONSE TIME testing. The response time assumptions of the CPCS Upgrade will be validated as part of the Site Acceptance Testing. Appendix B to the LTR provides the justification for this change.
TS 3.3.1 SR 4.3.1.4	SR 4.3.1.4 is no longer applicable, due to design changes, since isolation amplifiers and optical isolators are being replaced with fiber optic cabling which is qualified by Entergy, as described in LTR Section 6.2.2.19. The text of the SR is replaced with "DELETED".
TS 3.3.1 SR 4.3.1.5	SR 4.3.1.5 is no longer applicable since the upgraded CPCS design, using the Common Q platform, does not include the auto restart feature. The text of the SR is replaced with "DELETED".
TS 3.3.1 SR 4.3.1.6	SR 4.3.1.6 to perform a CHANNEL FUNCTIONAL TEST within 12 hours of receipt of a High CPC Cabinet Temperature alarm is being deleted. The basis for the removal of this SR is provided in Appendix B to the LTR

	and is consistent with the safety evaluation presented in Reference 10 and summarized below. The requirement to perform testing upon receipt of a cabinet high temperature alarm is not necessary and does not meet the criteria provided in 10 CFR 50.36(c)(2)(i) for demonstration of "lowest functional capability or performance levels of equipment required for safe operation of the facility." This is based on: a. A high CPC cabinet temperature alarm does not indicate the lowest functional capability or performance level of a CPC or CEAC. These alarms (122 deg F) are actuated well below the qualification temperature of the CPCs and CEAC (140 deg F) and merely inform the Operations staff of a potential challenge to CPC/CEAC operability. Typically, only one of four channels is affected on high cabinet temperature since each cabinet has its own independent cooling system. b. The existing SR requirement has no follow up requirements for continuous monitoring after the initial test to determine if functionality may be affected in the future with an existing high temperature condition. In contrast, the improved Common Q CPCS provides more extensive online diagnostics than the current CPCS and will continuously monitor and assess CPC/CEAC module functionality. These diagnostics address numerous failure conditions from many causes, temperature stress being only one such cause. Failures are flagged by pertinent error messages and a channel trouble alarm on the Operators Module (OM), Maintenance Test Panel (MTP) and remote annunciation. The improved CPCS design provides greater confidence in identifying and alarming on an actual loss of CPC/CEAC functionality. c. Lastly, the existence of a high CPC cabinet temperature alarm does not directly relate to when the CPCS becomes inoperable. Recognizing that upon receipt of the high temperature alarm, the operators have an annunciator response procedure to assess the condition and respond appropriately. The new cabinet RTDs will be periodically calibrated per the site’s calibration procedures.

TS 3.3.1 SR 4.3.1.7	SR 4.3.1.7 is being added to perform a test on the CPC DNBR/LPD trip output contact interface to the PPS. As described in LTR Appendix B, this portion of the system does not get monitored by the CPCS self-diagnostics. The test will be performed at the frequency prescribed in the Surveillance Frequency Control Program.
TS 3.3.1 Table 4.3-1	Table 4.3-1 is being changed to be consistent with the Functional Unit formatting changes described above for Tables 2.2-1 and 3.3-1, where the Core Protection Calculators are the designated Functional Unit 9, with Local Power Density – High, DNBR – Low, and CEACs listed as sub-functional units. The second change is that all entries for CHANNEL FUNCTIONAL TEST for all of the Functional Unit 9 lines are changed to "None." LTR Appendix B, along with the Waterford System Engineer and Operations Actions Supporting TS SR Reduction, as described in Section 3.4 below, provide the detailed justification that demonstrates that the self-diagnostics meet the requirements of 10 CFR 50.36 for the CPCS, except for the CPC DNBR/LPD trip output contacts which will be tested by the new SR 4.3.1.7. See also Section 3.4 below for Operations and site engineering actions. Table Notations (6) and (9) which describe elements of the CHANNEL FUNCTIONAL TEST are replaced with "DELETED". The verification described in notation (9) is incorporated in the design of the upgraded CPCS as described in LTR Appendix B, P.B-41, Item 1.
TS 3.10.2 SR 4.10.2.2	TS 3.10.2 is being revised in four places to replace "Functional Unit 15" with "Functional Unit 9c." This is purely editorial as a result of the changes to TS 2.2.1 and 3.3.1 described above, which redesignated the CPCs as Functional Unit 9c in Tables 2.2-1 and 3.3-1. SR 4.10.2.2 is being revised to replace "Functional Unit 15" with "Functional Unit 9c".

TS 6.8.1	Administrative TS 6.8.1 (g) is being revised to conform to specification 5.4.1.f of NUREG-1432 Revision 4, "Standard Technical Specifications – Combustion Engineering Plants". This change replaces the governing source document for modifications to the CPC software to the appropriate Common Q Software Program Manual and provides more substantive guidance for the control of CPC Type 1 addressable constants than the current site-specific guidance.
TS 6.9.1.11.1	Administrative TS 6.9.1.11.1 is being revised to conform with other proposed TS changes.

Attachment 2 contains the Clean TS pages reflecting incorporation of the changes described above.

3. TECHNICAL EVALUATION

The LAR is intended to address all of the DI&C-ISG-06 (Reference 2) content requirements for the Alternate Review Process (ARP). Enclosure B to DI&C-ISG-06, *Information Provided in Support of a License Amendment Request for a Digital Instrumentation and Control Modification*, provides a cross-reference to the descriptive material identified in the body of the DI&C-ISG-06 guidance document. This LAR addresses, as a minimum, items included in the Enclosure B "AR" column.

1. DI&C-ISG-06 Alternate Review Process (ARP) LAR Contents

DI&C-ISG-06 Section C.2 describes the ARP. Section C.2.1 provides guidance for ARP LAR contents. A prerequisite for requesting LAR review using the ARP is to use digital equipment which has a topical report previously approved by the NRC. There is also an expectation that the topical report vendor will develop the system. For the CPCS replacement, Entergy is proposing to use the Westinghouse Common Q digital platform. This platform has two NRC-approved topical reports for the application software development and for the digital equipment (References 7 and 11, respectively). The digital equipment topical report was recently re-reviewed by the NRC with an approval issued in January 2020. Thus, the equipment proposed for Waterford has been recently reviewed by the NRC. Note that LTR Section 6 (Attachment 4), which addresses DI&C-ISG-06 Section D.5, describes any differences between the Waterford system and that which is described in the NRC-approved topical reports. Westinghouse is contracted to develop the hardware and software system. The LTR addresses all of the Plant-Specific Action Items (PSAIs) and the remaining Generic Open Items (GOI) included in the most recent NRC approval for both topical reports.

There is a precedent for the CPCS design at Palo Verde Nuclear Generating Station Units 1, 2 and 3 (PVNGS). This is referred to as the reference design in the LAR and is described in LAR Section 4.2 below. The LTR describes the portions of the Waterford design that are similar to the PVNGS and have been previously reviewed by the NRC. The LAR includes the CPCS replacement project System Requirements Specification (SyRS) and Waterford-specific Failure Modes and Effects Analysis (FMEA). The SyRS project document has a reference design document (Attachment 7) and a "delta" document (Attachment 8) which describes differences for the Waterford project." Note that Revision 7 of the reference design SyRS (ADAMS Accession No. ML032830027) was previously reviewed by the NRC.

This is the pilot LAR for the ARP, and as such, this LAR is the first time a licensee has assembled the LAR content based on the DI&C-ISG-06 Revision 2 guidance. The ARP LAR is designed to be a single submittal provided to the NRC early in the project schedule. Thus, the LAR content is based on conceptual design, system requirements, and human-system interface requirements. Based on multiple NRC presubmittal meetings, Entergy believes the LAR contains sufficient "system design" information to demonstrate compliance with the regulatory requirements.

Both in DI&C-ISG-06 and in the public meetings held during its development, the NRC stressed the importance of licensees performing adequate vendor oversight of the digital platform vendor. The licensee has the primary responsibility to ensure that the vendor adheres to the lifecycle development process described in the LAR, NRC-approved vendor topical reports, and other procurement information. Waterford has developed a Vendor Oversight Plan (VOP) to ensure Westinghouse compliance to the NRC-approved development process and other procurement information. The VOP, as currently executed, is used to ensure that the vendor executes the project consistent with the LAR. A summary of the project-specific VOP is included in LAR Enclosure, Attachment 14.

Licensee Prerequisites

DI&C-ISG-06 Section C.2.2 describes the licensee prerequisites for use of the ARP. Item 1 states that the LAR should include a description of the licensee's VOP. The VOP, when executed must ensure that the vendor (1) executes the project consistent with the LAR, and (2) uses an adequate software QA program. As described above, the VOP summary is included in LAR Enclosure Attachment 14. The VOP describes the licensee interactions with the vendor throughout the entire system development lifecycle to ensure the software and system development is in accordance with the NRC-approved software development process (Reference 7).

Section C.2.2 Item 2 states that the LAR should contain a reference to an NRC-approved topical report. Item 2 has two subparts. To address subpart a. the Westinghouse Common Q platform has two NRC-approved topical reports (References 7 and 11). The CPCS application is within the scope of both topical reports. To address subpart b. Westinghouse will be using the NRC-approved Common Q Software Program Manual (SPM) (Reference 7) as the framework for the design and development of the Waterford CPCS replacement. This framework is a supplement to the Westinghouse 10 CFR 50 Appendix B Quality Assurance program to specifically address digital I&C safety system development.

Section C.2.2 Item 3 addresses licensee regulatory commitments (Attachment 15). This item has two subparts. Subpart a. states that the LAR should include regulatory commitments to complete the referenced topical reports' PSAIs. The LTR Sections 5 and 6 address the applicable PSAIs. In many instances, the PSAI response references vendor oversight. Through this LAR, Waterford will execute vendor oversight in accordance with the VOP. Based on one PSAI disposition, there is one regulatory commitment described in the Attachment 15 (i.e., SPM PSAI 5). Subpart b. states that the LAR should include regulatory commitments to complete lifecycle activities under the licensee's QA program similar to the activities a licensee would complete under a Tier 1, 2 or 3 licensing review. These activities are generically described in DI&C-ISG-06 Enclosure B. Based on an evaluation of the design activities completed at the time of LAR submittal and the activities covered by the VOP, no additional regulatory commitments are required.

2. Licensing Technical Report (LTR)

The LTR (Attachment 4) provides most of the LAR technical content. The LTR directly addresses the DI&C-ISG-06 Sections D.1 to D.8 subsections entitled "Information To Be Provided," which is delineated in the LTR Table of Contents. The various major section headings include "(D.x)". This parenthetical remark refers to the specific DI&C-ISG-06 sections with the x replaced with 1 to 8. Each section includes a description of compliance to the 10 CFR 50 Appendix A General Design Criteria or IEEE Std. 603 clauses or other regulatory requirements listed in the corresponding DI&C-ISG-06 section.

LTR Section 3.2.18 describes the NRC evaluation of the first CPCS at Arkansas Nuclear One, Unit 2 (ANO-2) in NUREG-0308, "Safety Evaluation Report Related to the Operation of Arkansas Nuclear One, Unit 2," Supplement 1 (i.e., the ANO-2 NRC SER) in regards to CPCS Common Cause Failure (CCF). This was also the evaluation the NRC staff referred to in their PVNGS safety evaluation for the Common Q CPCS upgrade license amendment (Reference 6.10, Section 3.4.6.11). The NRC cited the ANO-2 evaluation to conclude, in part, that CCF is adequately addressed for the Common Q CPCS replacement for PVNGS. The Waterford LTR included this as part of the reference design licensing precedence.

Waterford was licensed with a digital CPCS. The Waterford licensing basis for a postulated CPCS failure to trip due to a CCF is bounded by the Waterford 3 Anticipated Trip Without Scram (ATWS) Mitigation Systems described in FSAR Chapter 7.8. The ATWS mitigation systems are designed to mitigate the consequences of Anticipated Operational Occurrence (AOO's) coupled with a failure of the Reactor Protection System to trip the reactor.

There are two scenarios that could prevent the CPCS trips from completing the function of shutting down the reactor. The first scenario assumes the CPCS initiates the trip signal to mitigate an AOO but the analog PPS fails to complete the shutdown action after receiving the trip signal. This is the basic assumption for the ATWS mitigation systems.

The second scenario is the CPCS has a CCF failure that fails to send the trip signal to the analog PPS. This outcome is identical to the first scenario and therefore is bounded by the design of the ATWS mitigation systems.

The failure of the CEAs to insert to produce reactor shutdown during an AOO (i.e., an ATWS event) is the same scenario as a postulated CCF CPCS failure to initiate a trip for an AOO, as both result in the same plant response (i.e., CEAs fail to insert to produce a reactor shutdown when a CPC trip is expected for an AOO).

The CPCS, at the time the Waterford operating license was granted by the NRC, was, and remains, a digital computer system. The replacement Common Q CPCS is also a digital system, with all functions replicated with additional alarming and redundancy for greater reliability. Therefore, this "digital-to-digital" plant modification does not impact the design basis in FSAR Chapter 7.8. The same ATWS mitigation systems will be effective in protecting the health and safety of the public if the CPCS fails to trip due to a CCF.

In summary, the defense-in-depth and diversity licensing basis for Waterford is not adversely impacted by this modification.

LTR Section 7 provides a compliance matrix describing LAR compliance to IEEE Std. 603-1991 and IEEE Std. 7-4.3.2-2003 (References 12 and 15). The compliance matrix is based on the DI&C-ISG-06 example Table D-1, *IEEE Standards 603-1991 and 7-4.3.2-2003 Compliance/Conformance Tabl*e.

LTR Appendix A contains draft FSAR markups. These markups are being provided for information only in support of the LAR review. Entergy engineering procedures will govern FSAR revisions as a result of LAR approval and equipment installation. NRC will receive the Waterford updated FSAR as part of the biennial submittal per 10 CFR 50.71.

LTR Appendix B provides the Failure Modes, Effects, Diagnostics Analysis (FMEDA) and other analyses to support TS SR elimination. This appendix addresses the NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) Chapter 7 Branch Technical Position (BTP) 7-17, "Guidance on Self-Test and Surveillance Test Provisions," on self-test and surveillance test provisions. While the DI&C-ISG-06 Enclosure B AR column does not include a requirement for LAR inclusion of a FMEA, the Waterford-specific FMEA is included with the LAR (Attachment 10). This FMEA is included to support review of the Appendix B FMEDA for TS SR elimination. The Waterford-specific FMEA is considered a "living document" per DI&C-ISG-06.

LTR Appendix C includes Endnotes providing references (e.g., Entergy documents, Westinghouse documents, etc.) for statements of fact within the LTR.

3. <u>Factory Acceptance Test/Site Acceptance Test (FAT/SAT) Description</u>

While not required by the DI&C-ISG-06 ARP content requirements, the NRC safety evaluation (SE) for the PVNGS precedent (Reference 10) describes the NRC's review of testing as part of the acceptability of the application-specific software. Since the conduct of the FAT and SAT are outside of the LAR review scope, the following description is included to provide NRC assurance of adequate testing.

Based on the Software Program Manual (SPM) for Common Q™ Systems (Reference 7), the purpose of the Factory Acceptance Test (FAT) is to demonstrate that the complete system is integrated and functional. The FAT will be conducted at the Westinghouse facilities prior to shipment of equipment to Waterford. The FAT will be performed as a manufacturing test to provide evidence that the system meets its requirements and provides confidence that the site installation and integration activities will be successful. The FAT test, together with the documentation of the prior Verification and Validation (V&V) activities (module tests, unit tests, software code reviews, integration testing, and system validation testing, etc.) demonstrate full compliance to the requirements.

The FAT will contain a comprehensive suite of tests that cover the Waterford-specific System Requirements Specification (SyRS) (Reference 8) functional requirements. The completeness of the FAT is demonstrated by:

- Waterford-specific system tests performed
- Reference design system validation testing, performed previously, that remains valid for those design aspects that are identical

- Waterford design system testing is based on regression analysis per the SPM and testing requirements are validated as part of the independent V&V and confirmed by Vendor Oversight Plan (VOP) (Reference 9) audits
- The minimum set of tests required for a FAT defined in the Common Q SPM, Exhibit 7-1

The FAT is performed to:

- Demonstrate that the system being delivered has been manufactured correctly
- Demonstrate (in conjunction with V&V) compliance to requirements for customer acceptance
- Reduce the risk associated with deferring compliance demonstration to the site activities (e.g., SAT, preoperational testing, etc.)
- Demonstrate aspects of the design that would not be practical once full integration is achieved due to limitations on interfaces that are connected in the plant.

For design changes introduced for the Waterford system, regression analysis shall be performed to determine what tests need to be repeated or introduced to maintain the level of system design validation achieved during the first of a kind system validation test program. The system validation tests required by the regression analysis may be performed on the deliverable equipment as a separate section of the FAT or performed on surrogate equipment consistent with the regression testing methods. These methods have been reviewed and approved by the NRC as part of the Westinghouse SPM, as confirmed by the VOP audits.

The following test items shall be included or demonstrated in the FAT:

- Safety Functions
- Communications
- Operability of Displays
- Diagnostics associated with hardware specific inputs (door alarms, temperature alarms, breaker status, etc.)
- Performance (accuracy, time response, etc.)

The Waterford CPCS FAT will include overall functional testing for the Single Channel components and the Four Channel components. In addition to the Single Channel FAT and the Four Channel FAT, there are IV&V design verification tests conducted at the module and unit level for the Waterford CPCS changes. The Four Channel FAT will include IV&V software and all associated connections.

The FAT results will be comprised of multiple reports that will include test anomalies and failures that were dispositioned and corrected. Waterford project team personnel will observe the FAT under the VOP.

The Site Acceptance Test (SAT) is considered a two-part test verifying correct functionality and performance after the system is installed at Waterford. The Waterford Common Q CPCS site acceptance testing will be performed and controlled in accordance with Entergy procedures, and include pre-installation and post-installation tests. The pre-installation testing will include the tests considered to be the Site Acceptance Test (SAT). The primary intent of the SAT shall be to validate that the equipment was not damaged during shipment.

The SAT will include pre-installation testing at Waterford in the test area for the Single Channel and the Four Channel components after they are received on site. The Waterford test area will conform to all required Waterford procedures for software control and cyber security as part of a secure and controlled access area. The SAT is a reperformance of the applicable portions of the FAT, as part of receipt inspection, to ensure the full functionality of the delivered system. The SAT test procedures and reports are not complete at this time. The SAT is scheduled to be performed in March 2021 for the Single Channel, and November 2021 for the Four Channel systems. Westinghouse personnel are expected to be present during the SAT (in a support role only). Prior to performing the SAT, construction tests will be performed prior to initializing the CPC/CEAC system in the test area. Construction will include point to point (or scheme) checks, power and grounding checks, and an initial power-up check. After the SAT items are complete, dry runs of the CPCS post-installation tests will be performed to identify and correct any problems prior to the actual operability testing of the CPCS post-installation.

The post-installation tests will be conducted in two phases: post installation tests prior to declaring the CPCS operational, and tests to be performed after the CPCS is operational and the reactor is at power. Westinghouse personnel are expected to be present during the post-installation testing (in a support role only). The primary intent of the post-installation tests is to validate that the equipment was not damaged during installation and installed per the approved modification package. External system interface testing will be specified in the post-installation testing.

The post-installation tests prior to declaring the CPCS operable will include construction testing and functional testing. Construction tests will include point to point (or scheme) checks, power and grounding checks, and an initial power-up check. Functional testing will include annunciator operability, time response testing, channel calibration testing, channel interface testing, and system integrated tests.

The post-installation tests performed after declaring the CPCS operable are to ensure CEA movement, gather performance data, provide new baseline data, and to validate assumptions.

4. Waterford System Engineer and Operations Actions Supporting TS SR Reduction

As described in LTR Appendix B, the methodology to eliminate TS SRs leverages a precedent licensing action. Southern Nuclear Company (SNC) Vogtle Electric Generation Plant (VEGP) Units 3 and 4 requested a license amendment to eliminate a number of protection system TS SRs (Reference 13). The NRC approved this LAR in Reference 14. The NRC SE approved the removal of surveillance requirements related to the VEGP Units 3 and 4 Common Q-based safety system (i.e., the Protection and Safety Monitoring System (PMS)). As part of the NRC SE, the NRC described that

> "…plant administrative controls will be implemented to assure continued monitoring of the PMS system to assure adequate operation of the system diagnostic function. In the absence of the either divisional or system alarms, there will also be operator rounds and system engineer's monthly reports that evaluate and document the health, errors, and faults of system."

Note that while the LTR Appendix B states that monitoring is not required in order to credit self-diagnostic features, Waterford has elected to utilize operator rounds and system engineer activities to provide additional assurance that diagnostic faults are detected.

Post installation, CPCS operability will be verified using 1) the automated diagnostics credited in this LAR (i.e., as described in LTR Appendix B), 2) Technical Requirements Manual (TRM) 3/4.3.1, "Reactor Protective Instrumentation" and associated surveillance procedures; and 3) Waterford TS 6.5.1.8, "Surveillance Frequency Control Program (SFCP). A failure of credited automated diagnostics to detect a fault will be either detected by other diagnostics in the system or by checker(s) of diagnostics. This condition will be alarmed and displayed on the main control room (MCR) operator modules (OM) and/or the main control room annunciators. Upon receipt of an alarm or abnormal conditions, the station operating procedures will require the operators to perform system checks and verify operability of the CPCS deviation / function. The procedure will direct the operator to dispatch a maintenance technician to determine the source of the alarm as needed.

Procedure changes made as part of the implementation will impact routine Operations and site engineering actions. The following actions will also provide assurance (defense-in-depth) that diagnostic faults are captured and investigated.

1. Conduct of Operations, Operations Shift Logs, and Control Room Shift Logs – During routine operator rounds and MCR activities, the following tasks will be performed by the operators:
   - Checking the OMs for Health Status, alarms and faults
   - Checking the OMs CPCS Channel System Event Log, as described in the LTR Section 3.2.7.2.4 (Attachment 4), for Health Status, alarms and faults
   - Checking the OMs for failed sensor stack
   - Checking MCR annunciators

   The walkdowns and operator rounds are controlled by the plant procedures and the results are logged in accordance with plant procedures, which are continuously maintained and retrievable.

2. System Health Checks – Site engineers are required to establish and perform periodic system health monitoring and generate system health reports per Entergy procedures. Corrective actions are used to improve system health and the overall plant performance, safety and reliability. CPCS is a critical system which requires periodic system health monitoring and walk-down of the system. The CPC system checks include the following:

   - Failure trending of sub-components on CPC and CEAC circuit boards (as required)
   - CPC System Performance Indicator (PI) Trends – input instrument drift, sensor failures, system trips reviewed (various periodicities)
   - Review of trend data for CEAs including RSPTs and RSPT power supplies (weekly)
   - Walk-downs of the CPC system (quarterly)

System health reports are reviewed by systems engineering management and fleet subject matter experts. Required documentation, long range planning, and trending instruction are maintained in system notebooks. Issues are communicated, and adverse trends and issues not previously addressed (i.e., all alarms should have been addressed by Operations) are captured in condition reports.

4. REGULATORY EVALUATION

The Core Protection Calculator System (CPCS) replacement incorporates the fundamental design principles of redundancy, independence, deterministic behavior, and defense-in-depth and diversity while providing enhanced reliability and obsolescence management. The hardware and software development for the CPCS replacement complies with the Institute of Electrical and Electronics Engineers (IEEE) Standard 603-1991 Clause 5.3 "Quality," and IEEE Standard 7-4.3.2-2003 Clause 5.3 "Quality," including the digital system development life cycle, in order to provide a high quality development process (References 12 and 15). The independent V&V effort for the replacement utilizes a process that complies with IEEE Standard 7-4.3.2-2003 Clause 5.3.3, "Validation and Verification" to ensure the replacement meets the specified functional requirements and criteria.

Therefore, Entergy concludes the proposed CPCS replacement project complies with the 10 CFR 50 regulations and associated regulatory guidance.

1. Applicable Regulatory Requirements/Criteria

The following regulations and guidance are applicable to the proposed CPCS replacement project installation:

- 10 CFR 50.36, "Technical Specifications." The criteria for limiting conditions for operation and surveillance requirements are in 50.36(c)(2) and (3), respectively.

- 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

- Paragraph 10 CFR 50.54(jj), "Conditions of Licenses," states that structures, systems, and components subject to the codes and standards of 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

- Paragraph 10 CFR 50.55a(h), "Protection and safety systems," approves the 1991 version of IEEE Standard 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," for incorporation by reference including the correction sheet dated January 30, 1995.

- Paragraph 10 CFR 50.55(i), " Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses," states that structures, systems, and components subject to the codes and standards of 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

- The following General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 are addressed in the LTR (Attachment 4):

    - GDC 1, "Quality Standards and Records"
    - GDC 13, "Instrumentation and control"
    - GDC 21, "Protection system reliability and testability"
    - GDC 22, "Protective system independence"
    - GDC 23, "Protection system failure modes"
    - GDC 24, "Separation of protection and control systems"
    - GDC 29, "Protection against anticipated operational occurrences"

- 10 CFR 50, Appendix A, GDC 4, "Environmental and dynamic effects design bases," requires that the core protection calculator system (CPCS) be designed and qualified to operate under the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. The protection system shall also be appropriately protected against dynamic effects. The CPCS equipment qualification is contained in the Equipment Qualification Summary Report (EQSR) which is referenced in WCAP-18484-P, "Licensing Technical Report for the Waterford Steam Electric Station Unit 3 Common Q Core Protection Calculator System," Section 4, "Hardware Equipment Qualification (D.3)."

- 10 CFR 50, Appendix A, GDC 10, "Reactor design," requires that specified acceptable fuel design limits (SAFDLs) are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis (i.e., a 95/95 probability/confidence level criteria) that DNB will not occur on the limiting fuel rods, and by requiring that fuel centerline temperature stays below the melting temperature. The reactor core safety limits are established to preclude violation of these criteria. Automatic enforcement of the reactor core safety limits is provided by the reactor protection system (RPS), which includes a number of reactor trip functions, two of which are the DNBR - low and local power density (LPD) - high reactor trips. As part of the RPS, the CPCS generates a reactor trip signal when the DNBR or the LPD approach their specified limiting safety system settings. The reactor trips protect against violating core SAFDLs during AOO's. In meeting GDC 10, the replacement CPCS continues to satisfy these functional requirements.

- 10 CFR 50, Appendix A, GDC 20, "Protection system functions," requires that protection system functions shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. The CPCS is designed to meet this GDC requirement. The Waterford design basis functions of the CPCS are unchanged as a result of this modification to the Common Q CPCS. These same CPCS design basis functions are found in the Common Q CPCS reference design (i.e., PVNGS CPCS).

- 10 CFR 50, Appendix A, GDC 25, "Protection system requirements for reactivity control malfunctions," provides protection system requirements for reactivity control malfunctions. It states that the protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods. The CPCS is designed to mitigate reactivity malfunctions as described in the Waterford FSAR, Chapter 15. The Waterford design basis functions of the CPCS are unchanged as a result of this modification to the Common Q CPCS. These same CPCS design basis functions are found in the Common Q CPCS reference design (i.e., PVNGS CPCS).

- Regulatory Guide 1.53, Revision 2, "Application of the Single-Failure Criterion to Safety Systems," November 2003 (ADAMS Accession No. ML033220006).

- Regulatory Guide 1.75, Revision 3, "Physical Independence of Electric Systems," February 2005 (ADAMS Accession No. ML13350A340).

- Regulatory Guide 1.89, Revision 1, " Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," June 1984 (ADAMS Accession No. ML003740271).

- Regulatory Guide 1.100, Revision 3, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," September 2009 (ADAMS Accession No. ML091320468).

- Regulatory Guide 1.152, Revision 3, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," July 2011 (ADAMS Accession No. ML102870022).

- Regulatory Guide 1.170, Revision 1, "Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," July 2013 (ADAMS Accession No. ML13003A216).

- Regulatory Guide 1.172, Revision 1 "Software Requirements Specifications for Digital Computer Software and Complex Electronics Used in Safety Systems of Nuclear Power Plants," July 2013 (ADAMS Accession No. ML13007A173).

- Regulatory Guide 1.180, Revision 1, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," October 2003 (ADAMS Accession No. ML032740277).

- Regulatory Guide 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants," March 2007 (ADAMS Accession No. ML070190294).

- NUREG-0711, Revision 3, "Human Factors Engineering Program Review Model," November 2012 (ADAMS Accession No. ML12324A013).

- NUREG-1764, Revision 1, "Guidance for the Review of Changes to Human Actions," September 2007 (ADAMS Accession No. ML072640413).

- DI&C-ISG-04, Revision 1, "Task Working Group #4: Highly-Integrated Control Rooms- Communications Issues (HICRc)," March 2007 (ADAMS Accession No. ML083310185).

- DI&C-ISG-06, "Task Working Group #6: Licensing Process," Revision 2, dated December 2018 (ADAMS Accession No. ML18269A259).

- The applicable portions of the following branch technical positions within NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Chapter 7, "Instrumentation and Controls," as follows:

  - Branch Technical Position 7-14, "Guidance on Software Reviews for Digital Computer- Based Instrumentation and Control Systems"

  - Branch Technical Position 7-17, " Guidance on Self-Test and Surveillance Test Provisions"

The Licensing Technical Report (Attachment 4) and other attachments contain project-specific compliance information for the above regulations and guidance.

2. Precedent

The Core Protection Calculator System (CPCS) at Waterford Steam Electric Station Unit 3 (Waterford) is being replaced with a new system based on the Common Qualified (Common Q™) Platform. This system is based on the reference design that is installed at the Palo Verde Nuclear Generating Station Units 1, 2, and 3 (PVNGS) that was reviewed and approved by the NRC (Reference 10). There are minor architectural changes from the reference design as a result of obsolescence and unique aspects of the Waterford plant compared to PVNGS. The Licensing Technical Report (Attachment 4) describes the Waterford Common Q CPCS and identifies where the implementation is the same as PVNGS to assist the NRC staff in their review of the LAR.

3. No Significant Hazards Consideration Analysis

In accordance with 10 CFR 50.90, Entergy Operations, Inc. (Entergy) requests an amendment to Renewed Facility Operating License (FOL) No. NPF-38, Appendix A, "Technical Specifications" (TSs) for Waterford Steam Electric Station, Unit 3 (Waterford). The proposed TS changes reflect the upgrade of the Waterford digital Core Protection Calculator System (CPCS), comprised of CPCs and Control Element Assembly Calculators (CEACs), with a new, more reliable digital system based on the NRC-approved Westinghouse Common Qualified (Common Q™) Platform.

The following Technical Specifications (TS) sections are affected by this change:

2.2.1	Reactor Trip Setpoints
3.1.3.1	CEA Position

3.2.4 DNBR Margin
3.3.1 Reactor Protective Instrumentation
3.10.2 Moderator Temperature Coefficient, Group Height, Insertion, and Power Distribution Limits
6.8.1 Procedures and Programs
6.9 Reporting Requirements

Entergy has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three conditions set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The probability of accidents occurring is not affected by the proposed amendment. The CPCS is not the initiator of any accident and does not interact with equipment whose failure could cause an accident. The CPCS provides reactor trips to the Reactor Protection System (RPS) for high local power density (LPD) and low departure from nucleate boiling ratio (DNBR). All design basis events, and the reliance on the CPCS low DNBR and high LPD trips will remain unchanged.

   The consequences of accidents are not affected by the proposed amendment. The upgrade of the CPCS will change the existing system architecture in the area of CEAC processing by transitioning from two CEACs (i.e., providing input to four CPC channels) to eight CEACs (i.e., two in each CPC channel). Increasing the number of CEACs to eight will increase the availability of the CEAC processing. The CPCS functional design and design basis functions will not change as a result of the proposed amendment.

   The availability of the upgraded CPCS system will be equal to or greater than the existing system and, as a result, the scram reliability will be equal to or better than the existing system. The requirements for response time and accuracy that are assumed in the Waterford Updated Final Safety Analysis Report (UFSAR) accident analysis will continue to be met. Therefore, the new CPCS will be capable of performing the same safety-related functions within the same response time and accuracy as the existing CPCS. No new challenges to safety-related equipment will result from the CPCS modification. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The functional design of the CPCS system and the design basis functions will not change as a result of the CPCS modification. The components of the CPCS will be supplied to equivalent or better design and qualification criteria than is currently required for Waterford. The CPCS modification will not introduce any new operating modes, safety-related equipment lineups, accident scenarios, system interactions, or failure modes that would create a new or different type of accident. Failure(s) of the system will have the same overall effect as the present design. Therefore, the upgraded CPCS will not adversely affect plant equipment.

   The existing CPCS is implemented in computer-based hardware, therefore implementation of the NRC-approved Westinghouse Common Q™ platform represents a digital-to-digital upgrade. The original licensing basis for Waterford assumes a potential common cause failure of the CPCS. The replacement of the current digital CPCS with the Common Q™ platform does not change the Waterford licensing basis for defense-in-depth and diversity. Therefore, the proposed change does not result in any new common cause failure or any reduction in defense-in-depth and diversity.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed TS changes associated with the CPCS modification implement the constraints associated with eight CEACs, relative to the current design of two CEACs. This new design, as well as the implementation of the Westinghouse Common Q™ platform does not impact reactor operating parameters or the functional requirements of the CPCS. The CPCS will continue to provide reactor trips to the RPS for high LPD and low DNBR. All design basis events, and the reliance on the CPCS low DNBR and high LPD trips will remain unchanged.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety. Based on the above, Entergy concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4. Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6. REFERENCES

6.1 Entergy Operations, Inc. (Entergy) letter to U.S. Nuclear Regulatory Commission (NRC), "Letter-of-Intent to Submit License Amendment Requesting Using Digital Instrumentation and Control Interim Staff Guidance-(ISG)-06, Revision 2 and Request for NRC Fee Waiver," (ADAMS Accession No. ML19137A082), dated May 16, 2019

6.2 U.S. NRC Digital Instrumentation and Control Interim Staff Guidance-(ISG)-06, "Licensing Process," Revision 2 (ADAMS Accession No. ML18269A259)

6.3 NUREG-1764, Guidance for the Review of Changes to Human Actions, Revision 1 (ADAMS Accession No. ML072640413), U.S. NRC

6.4 NUREG-0711, Human Factors Engineering Program Review Model, Revision 3, November 2012 (ADAMS Accession No. ML072640413), U.S. NRC

6.5 NUREG/CR-6400, Human Factors Engineering (HFE) Insights for Advanced Reactors Based Upon Operating Experience, January 1997, (ADAMS Accession No. ML072640413), U.S. NRC

6.6 Entergy-Westinghouse Contract #10575450-01 including Purchase Orders 10587546, 10591996

6.7 Software Program Manual for Common Q™ Systems, WCAP-16096-P-A, Revision 5, Westinghouse Electric Company LLC

6.8 System Requirements Specification for the Core Protection Calculator System, WNA-DS-04517-CWTR3, Revision 2, Westinghouse Electric Company LLC

6.9 Waterford Unit Steam and Electric Station Unit 3 Core Protection Calculator System Replacement Project Vendor Oversight Plan, Entergy document no. VOP-WF3-2019-00236, Revision 1

6.10 Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendments on the Core Protection Calculator System Upgrade (TAC Nos. MB6726, MB6727 and MB6728) (ADAMS Accession No. ML033030363), U.S. NRC

6.11 Common Qualified Platform Topical Report, WCAP-16097-P-A, Revision 4, Westinghouse Electric Company LLC

6.12 IEEE Standard 603-1991, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations

6.13 Vogtle Electric Generation Plant Units 3 and 4 – Request for Licenses Amendment Regarding Protection and Safety Monitoring System Surveillance Requirement Reduction Technical Specification Revision (LAR 19-001), (ADAMS Accession No. ML19084A309), Southern Nuclear Company

6.14 Vogtle Electric Generating Plant Units 3 and 4 Safety Evaluation (LAR 19-001), (ADAMS Accession No. ML19297D159), U.S. NRC

6.15 IEEE Standard 7-4.3.2-2003, IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations

7. ATTACHMENTS

1. Technical Specification Page Markups, Revision 1
2. Clean Technical Specification Pages, Revision 1

Enclosure 1, Attachment 1

W3F1-2021-0032

Technical Specification Page Mark-ups, Revision 1

TS Pages

2-3
3/4 1-20
3/4 2-6
3/4 3-1
3/4 3-2
3/4 3-3
3/4 3-4
3/4 3-6
3/4 3-7
3/4 3-10
3/4 3-11
3/4 3-12a
3/4 10-2
6-14
6-20
6-20a
7 Pages of INSERTS

TABLE 2.2-1
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
Four Reactor Coolant Pumps Operating	$\leq$ 108% of RATED THERMAL POWER	$\leq$ 108.76% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.257% of RATED THERMAL POWER (6)	$\leq$ 0.280% of RATED THERMAL POWER (6)
4. Pressurizer Pressure - High	$\leq$ 2350 psia	$\leq$ 2359 psia
5. Pressurizer Pressure - Low	$\geq$ 1684 psia (2)	$\geq$ 1649.7 psia (2)
6. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.4 psia
7. Steam Generator Pressure - Low	$\geq$ 666 psia (3)	$\geq$ 652.4 psia (3)
8. Steam Generator Level - Low	$\geq$ 27.4% (4)	$\geq$ 26.48% (4)
9. Local Power Density - High	$\leq$ 21.0 kW/ft (5)	$\leq$ 21.0 kW/ft (5)
10. DNBR - Low	$\geq$ 1.26 (5)	$\geq$ 1.26 (5)
11. DELETED		
12. Reactor Protection System Logic	Not Applicable	Not Applicable
13. Reactor Trip Breakers	Not Applicable	Not Applicable
14. Core Protection Calculators	Not Applicable	Not Applicable
15. CEA Calculators	Not Applicable	Not Applicable
16. Reactor Coolant Flow - Low	$\geq$ 19.00 psid (7)	$\geq$ 18.47 psid (7)

Replace with INSERT A

Delete and leave blank

Replace with "DELETED"

INSERT A

9. Core Protection Calculators
    a. Local Power Density - High
    b. DNBR – Low

10. DELETED

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group in accordance with the Surveillance Frequency Control Program except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

Delete

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction in accordance with the Surveillance Frequency Control Program.

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by one of the following methods:

Insert B

Replace with "1."

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both CEACs are operable); or

Replace with Insert C

Replace with "2."

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the amount specified in the COLR (when COLSS is in service and neither CEAC is operable); or

Replace with Insert D

Replace with "1."

c. Operating within the region of acceptable operation specified in the COLR using any operable CPC channel (when COLSS is out of service and either one or both CEACs are operable); or

Replace with Insert E

Replace with "2."

d. Operating within the region of acceptable operation specified in the COLR using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

Replace with Insert F

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With the DNBR limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

   1. Restore the DNBR to within its limits within 1 hour, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With the DNBR limit not being maintained as indicated by operation outside the region of acceptable operation specified in the COLR with COLSS out of service, either:

   1. Restore COLSS to service within 2 hours, or

   2. Restore the DNBR to within its limits within the next 2 hours, or

   3. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

INSERT B

a. Core Operating Limit Supervisory System (COLSS) in Service:

INSERT C

when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE Core Protection Calculator (CPC) Channel; or

INSERT D

when the CEAC requirements of LCO 3.2.4.a.1 are not met.

b. COLSS Out of Service

INSERT E

OPERABLE Core Protection Calculator (CPC) Channel when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE CPC channel; or

INSERT F

OPERABLE Core Protection Calculator (CPC) Channel (with both CEACS inoperable) when the CEAC requirements of LCO 3.2.4.b.1 are not met.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function such that all channels are tested as shown in the "Total No. of Channels" column of Table 3.3-1.

Replace with Insert G

4.3.1.4 The isolation characteristics of each CEA isolation amplifier and each optical isolator for CEA Calculator to Core Protection Calculator data transfer shall be verified in accordance with the Surveillance Frequency Control Program during the shutdown per the following tests:

a. For the CEA position isolation amplifiers:

1. With I20 volts AC (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not exceed 0.015 volts DC.

Replace with "DELETED"

INSERT G

Neutron detectors, Core Protection Calculators, and CEACs

SURVEILLANCE REQUIREMENTS (Continued)

2. With 120 volts AC (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15.0 volts DC.

b. For the optical isolators: Verify that the input to output insulation resistance is greater than 10 megohms when tested using a megohmmeter on the 500 volt DC range.

Delete

4.3.1.5 The Core Protection Calculator System and the Control Element Assembly Calculator System shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.6 The Core Protection Calculator System shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

Replace with "DELETED" and relocate to page 3/4 3-1

4.3.1.7 Perform a test on the CPC DNBR/LPD trip output through the contact interface to the PPS in accordance with the Surveillance Frequency Control Program.

Add new 4.3.1.7 and locate on page 3/4 3-1

THIS PAGE HAS BEEN DELETED

Add and center on page

TABLE 3.3-1
REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	8
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level-High					
a. Startup and Operating	4	2(a)(d)	3	2**	2#, 3#
	4	2	3	3*, 4*, 5*	8
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
11. DELETED					
12. Reactor Protection System Logic	4	2	3	1, 2	5
				3*, 4*, 5*	8
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 5*	8
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2#, 3# and 7
15. CEA Calculators	2	1	2(e)	1, 2	6 and 7
16. Reactor Coolant Flow - Low	4/SG	2/SG(c)	3/SG	1, 2	2#, 3#

Replace with Insert H

Replace with "DELETED"

INSERT H

9. Core Protection Calculators	4	2(c)(d)(h)	3	1,2	2#, 3#
a. Local Power Density - High					
b. DNBR – Low					
c. CEA Calculators	4 (g)(i)	2(e)	3 (g)(i)	1,2	6
10. DELETED					

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

**Not applicable above a logarithmic power of 10^{-4} % RATED THERMAL POWER.

(a) The operating bypass may be enabled above the 10^{-4}% bistable setpoint and shall be capable of automatic removal whenever the operating bypass is enabled and logarithmic power is below the 10^{-4}% bistable setpoint. Trip may be manually bypassed during physics testing pursuant to Special Test Exception 3.10.3.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) The operating bypass may be enabled below the 10^{-4}% bistable setpoint and shall be capable of automatic removal whenever the operating bypass is enabled and logarithmic power is above the 10^{-4}% bistable setpoint. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; the 10^{-4}% bistable setpoint may be changed to less than or equal 5% RATED THERMAL POWER to perform the automatic removal function.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) There are two CEACs in each CPC channel.

(h) Both Local Power Density- HIgh and DNBR-Low must be OPERABLE for a CPC Channel to be OPERABLE.

(i) Both CEACs in an inoperable CPC channel are also inoperable.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3. Containment Pressure - (RPS) High	Containment Pressure - High Containment Pressure - High (ESF)
4. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5. Steam Generator Level	Steam Generator Level - Low Steam Generator ΔP (EFAS)
6. Core Protection Calculator	Local Power Density - High DNBR - Low
7. Logarithmic Power	Logarithmic Power Level - High Local Power Density - High (1) DNBR - Low (1) Reactor Coolant Flow - Low (1)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes. *

ACTION 5 - With the number of channels OPERABLE one less those required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour; otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group. After 7 days, operation may continue provided that Actions 6.b.1, 6.b.2, and 6.b.3 are met.

* Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

(1) With the operating bypass enabled.

Replace with Insert I and Move Action 6 to page 3/4 3-7

INSERT I

ACTION 6 - Separate Actions may be entered for each CPC channel.

a. With one CEAC inoperable in 1 or 2 CPC channels, either declare the associated CPC channel(s) inoperable; or set the “RSPT/CEAC Inoperable” addressable constant to the inoperable status within 4 hours.

b. With one CEAC inoperable in 3 or 4 CPC channels, either declare the associated CPC channel(s) inoperable; or, operation may continue provided that:

1. Within 4 hours the “RSPT/CEAC Inoperable” addressable constant(s) is set to the inoperable status.

2. Operation may continue for up to 7 days provided that the position of each CEA is verified to be aligned with all other CEAs in its group by performing surveillance requirement 4.1.3.1.1 at least once per 4 hours.

3. Operation may continue after 7 days provided that Actions 6.c.1, 6.c.2, and 6.c.3 are met

c. With both CEACS inoperable in any CPC channel, either declare the associated CPC channel(s) inoperable; or, operation may continue provided that:

1. Within 4 hours the DNBR margin required by Specification 3.2.4a (COLSS in service) or 3.2.4b (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

2. Within 4 hours:

   a) All CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then perform surveillance requirement 4.1.3.1.1 at least once per 4 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

Delete

b. With both CEACs inoperable, operation may continue provided that:

1. Within 4 hours the DNBR margin required by Specification 3.2.4b (COLSS in service) or 3.2.4d (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

2. Within 4 hours:

   a) All CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

Replace with "DELETED" and move ACTIONS 7 & 8 to new page 3/4 3-7a

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers withi[n the] next hour.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	SFCP and S/U(1)	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	SFCP	SFCP(2,4),SFCP (3,4), SFCP (4)	SFCP	1, 2
3. Logarithmic Power Level - High	SFCP	SFCP(4)	SFCP and S/U(1)	2#, 3, 4, 5
4. Pressurizer Pressure - High	SFCP	SFCP	SFCP	1, 2
5. Pressurizer Pressure - Low	SFCP	SFCP	SFCP	1, 2
6. Containment Pressure - High	SFCP	SFCP	SFCP	1, 2
7. Steam Generator Pressure - Low	SFCP	SFCP	SFCP	1, 2
8. Steam Generator Level - Low	SFCP	SFCP	SFCP	1, 2
9. Local Power Density - High	SFCP	SFCP(2,4),SFCP(4,5)	SFCP, SFCP(6)	1, 2
10. DNBR - Low	SFCP	SFCP(7), SFCP(2,4), SFCP(8), SFCP(4,5)	SFCP, SFCP(6)	1, 2
11. DELETED				
12. Reactor Protection System Logic	N.A.	N.A.	SFCP(11) and S/U(1)	1, 2, 3*, 4*, 5*

Replace with INSERT J

INSERT J

9. Core Protection Calculators	SFCP	SFCP(2,4), SFCP(4,5)	None	1,2
a. Local Power Density – High	SFCP	SFCP(2,4), SFCP(4,5)	None	1,2
b. DNBR – Low	SFCP	SFCP(7), SFCP(2,4), SFCP(8), SFCP(4,5)	None	1,2
c. CEA Calculators	SFCP	SFCP	None	1,2
10. DELETED				

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	SFCP(10,11), S/U(1)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	SFCP	SFCP(2,4), SFCP(4,5)	SFCP(9), SFCP(6)	1,2
15. CEA Calculators	SFCP	SFCP	SFCP, SFCP(6)	1, 2
16. Reactor Coolant Flow - Low	SFCP	SFCP	SFCP	1, 2

Replace with "DELETED"

Delete and leave blank

(3) Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine or verify acceptable values for the shape annealing matrix elements used in the Core Protection Calculators.

(6) This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

Replace with "DELETED"

(7) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow co-efficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty is included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations.

(9) The CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC.

(10) In accordance with the Surveillance Frequency Control Program and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage trip function and the shunt trip function.

(11) The CHANNEL FUNCTIONAL TEST shall be scheduled and performed such that the Reactor Trip Breakers (RTBs) are tested at least every 6 weeks to accommodate the appropriate vendor recommended interval for cycling of each RTB.

SPECIAL TEST EXCEPTIONS

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

Replace "15" with "9c"

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detection Monitoring System pursuant to the requirements of Specifications 4.2.1.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

6.6 NOT USED

6.7 NOT USED

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and Emergency Operating Procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Not used.

e. Not used.

f. Not used.

Replace with INSERT K

g. Modification of Core Protection Calculator (CPC) Addressable Constants, including independent verification of modified constants.

NOTES:

(1) Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the On-Site Safety Review Committee.

(2) Modifications to the CPC software (including algorithm changes and changes in fuel cycle specific data) shall be performed in accordance with the most recent version of CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure," that has been determined to be applicable to the facility. Additions or deletions to CPC Addressable Constants or changes to Addressable Constant software limits values shall not be implemented without prior NRC approval.

h. Administrative procedures implementing the overtime guidelines of Specification 6.2.2e., including provisions for documentation of deviations.

i. PROCESS CONTROL PROGRAM implementation.

INSERT K

g. Modification of core protection calculator (CPC) addressable constants.

These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of WCAP-16096-P-A, "CPC Protection Algorithm Software Change Procedure," which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

INDUSTRIAL SURVEY OF TOXIC OR HAZARDOUS CHEMICALS REPORT

6.9.1.9 Surveys and analyses of major industries in the vicinity of Waterford 3 which could have significant inventories of toxic chemicals onsite to determine impact on safety shall be performed and submitted to the Commission at least once every 4 years.

6.9.1.10 A survey of major pipelines ($\geq$ 4 inches) within a 2-mile radius of Waterford 3, which contain explosive or flammable materials and may represent a hazard to Waterford 3, including scaled engineering drawings or maps which indicate the pipeline locations, shall be performed and submitted to the Commission at least once every 4 years.

CORE OPERATING LIMITS REPORT COLR

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or any remaining part of a reload cycle for the following:

3.1.1.1 SHUTDOWN MARGIN – ANY CEA WITHDRAWN
3.1.1.2 SHUTDOWN MARGIN – ALL CEAS FULLY INSERTED
3.1.1.3 MODERATOR TEMPERATURE COEFFICIENT
3.1.2.9 BORON DILUTION
3.1.3.1 CEA POSITION
3.1.3.6 REGULATING AND GROUP P CEA INSERTION LIMITS
3.2.1 LINEAR HEAT RATE
3.2.3 AZIMUTHAL POWER TILT – T_q
3.2.4 DNBR MARGIN
3.2.7 AXIAL SHAPE INDEX
3.6.1.5 AIR TEMPERATURE, CONTAINMENT (Linear Heat Rate, 3.2.1)
3.9.1 BORON CONCENTRATION

6.9.1.11.1 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as follows:

Replace "3.2.4.b" with "3.2.4.a.2"

1) "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" (WCAP-11596-P-A), "ANC: A Westinghouse Advanced Nodal Computer Code" (WCAP-10965-P-A), and "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery" (WCAP-10965-P-A Addendum 1) (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, 3.2.4.b for DNBR Margin, 3.1.2.9 for Boron Dilution, and 3.9.1 for Boron Concentrations).

2) "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A (Methodology for Specification 3.1.3.6 for Regulating and Group P CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

CORE OPERATING LIMITS REPORT COLR (Continued)

3) "Modified Statistical Combination of Uncertainties, CEN-356(V)-P-A, Revision 01-P-A (Methodology for Specification 3.2.4.c and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

Replace "3.2.4.c" with "3.2.4.b.1"

Replace "3.2.4.d" with "3.2.4.b.2"

4) "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

5) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6) "Technical Manual for the CENTS Code," WCAP-15996-P-A, Rev. 1 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.3 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

7) "Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

Replace "3.2.4.b" with "3.2.4.a.2"

8) "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (may be used as a replacement for the PHOENIX-P lattice code as the methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, 3.2.4.b for DNBR Margin, 3.1.2.9 for Boron Dilution, and 3.9.1 for Boron Concentrations).

9) "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," WCAP-16072-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Tilt, and 3.2.7 for ASI).

10) "CE 16 x 16 Next Generation Fuel Core Reference Report," WCAP-16500-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, 3.2.4.b, 3.2.4.c and 3.2.4.d for DNBR Margin, and 3.2.7 for ASI).

11) "Optimized ZIRLO™," WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

Replace with "3.2.4.a.2, 3.2.4.b.1 and 3.2.4.b.2"

12) "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," WCAP- 16523-P-A (Methodology for Specification 3.2.4.b, 3.2.4.c and 3.2.4.d for DNBR Margin).

13) "ABB Critical Heat Flux Correlations for PWR Fuel," CENPD-387-P-A (Methodology for Specification 3.2.4.b, 3.2.4.c and 3.2.4.d for DNBR Margin and 3.2.7 for ASI).

Enclosure 1, Attachment 2

W3F1-2021-0032

Clean Technical Specification Pages, Revision 1

TS Pages

2-3
3/4 1-20
3/4 2-6
3/4 3-1
3/4 3-2
3/4 3-3
3/4 3-4
3/4 3-6
3/4 3-7
3/4 3-7a (new)
3/4 3-10
3/4 3-11
3/4 3-12a
3/4 10-2
6-14
6-20
6-20a

TABLE 2.2-1
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
Four Reactor Coolant Pumps Operating	≤ 108% of RATED THERMAL POWER	≤ 108.76% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.257% of RATED THERMAL POWER (6)	≤ 0.280% of RATED THERMAL POWER (6)
4. Pressurizer Pressure - High	≤ 2350 psia	≤ 2359 psia
5. Pressurizer Pressure - Low	≥ 1684 psia (2)	≥ 1649.7 psia (2)
6. Containment Pressure - High	≤ 17.1 psia	≤ 17.4 psia
7. Steam Generator Pressure - Low	≥ 666 psia (3)	≥ 652.4 psia (3)
8. Steam Generator Level - Low	≥ 27.4% (4)	≥ 26.48% (4)
9. Core Protection Calculators		
a. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
b. DNBR - Low	≥ 1.26 (5)	≥ 1.26 (5)
10. DELETED		
11. DELETED		
12. Reactor Protection System Logic	Not Applicable	Not Applicable
13. Reactor Trip Breakers	Not Applicable	Not Applicable
14. DELETED		
15. DELETED		
16. Reactor Coolant Flow - Low	≥ 19.00 psid (7)	≥ 18.47 psid (7)

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group in accordance with the Surveillance Frequency Control Program.

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction in accordance with the Surveillance Frequency Control Program.

POWER DISTRIBUTION LIMITS

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by one of the following methods:

a. Core Operating Limit Supervisory System (COLSS) in Service:

1. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE Core Protection Calculator (CPC) channel; or

2. Maintaining COLSS calculated core power less than or equal to COLSS Calculated core power operating limit based on DNBR decreased by the amount specified in the COLR when the CEAC requirements of LCO 3.2.4.a.1 are not met.

b. COLSS Out of Service

1. Operating within the region of acceptable operation specified in the COLR using any OPERABLE Core Protection Calculator (CPC) channel when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE CPC channel; or

2. Operating within the region of acceptable operation specified in the COLR using any OPERABLE Core Protection Calculator (CPC) channel (with both CEACS inoperable) when the CEAC requirements of LCO 3.2.4.b.1 are not met.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With the DNBR limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

1. Restore the DNBR to within its limits within 1 hour, or

2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors, Core Protection Calculators, and CEACs are exempt from response time testing. Each test shall include at least one channel per function such that all channels are tested as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.4 DELETED

4.3.1.5 DELETED

4.3.1.6 DELETED

4.3.1.7 Perform a test on the CPC DNBR/LPD trip output through the contact interface to the PPS in accordance with the Surveillance Frequency Control Program.

THIS PAGE HAS BEEN DELETED.

TABLE 3.3-1
REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	8
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level-High					
a. Startup and Operating	4	2(a)(d)	3	2**	2#, 3#
	4	2	3	3*, 4*, 5* 8	
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level – Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Core Protection Calculators	4	2(c)(d)(h)	3	1, 2	2#, 3#
a. Local Power Density – High					
b. DNBR – Low					
c. CEA Calculators	4(g)(i)	2(e)	3(g)(i)	1, 2	6
10. DELETED					
11. DELETED					
12. Reactor Protection System Logic	4	2	3	1, 2	5
				3*, 4*, 5*	8
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 5*	8
14. DELETED					
15. DELETED					
16. Reactor Coolant Flow - Low	4/SG	2/SG(c)	3/SG	1, 2	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

**Not applicable above a logarithmic power of 10^{-4} % RATED THERMAL POWER.

(a) The operating bypass may be enabled above the 10^{-4}% bistable setpoint and shall be capable of automatic removal whenever the operating bypass is enabled and logarithmic power is below the 10^{-4}% bistable setpoint. Trip may be manually bypassed during physics testing pursuant to Special Test Exception 3.10.3.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) The operating bypass may be enabled below the 10^{-4}% bistable setpoint and shall be capable of automatic removal whenever the operating bypass is enabled and logarithmic power is above the 10^{-4}% bistable setpoint. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; the 10^{-4}% bistable setpoint may be changed to less than or equal 5% RATED THERMAL POWER to perform the automatic removal function.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) There are two CEACs in each CPC channel.

(h) Both Local Power Density-High and DNBR-Low must be OPERABLE for a CPC channel to be OPERABLE.

(i) Both CEACs in an inoperable CPC channel are also inoperable.

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - (RPS) High	Containment Pressure - High Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low
7.	Logarithmic Power	Logarithmic Power Level - High Local Power Density - High (1) DNBR - Low (1) Reactor Coolant Flow - Low (1)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes. *

ACTION 5 - With the number of channels OPERABLE one less those required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour; otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

* Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

(1) With the operating bypass enabled.

ACTION STATEMENTS

ACTION 6 - Separate Actions may be entered for each CPC channel.

a. With one CEAC inoperable in 1 or 2 CPC channels, either declare the associated CPC channel(s) inoperable; or set the “RSPT/CEAC Inoperable” addressable constant to the inoperable status within 4 hours.

b. With one CEAC inoperable in 3 or 4 CPC channels, either declare the associated CPC channel(s) inoperable; or, operation may continue provided that:

1. Within 4 hours the “RSPT/CEAC Inoperable” addressable constant(s) is set to the inoperable status.

2. Operation may continue for up to 7 days provided that the position of each CEA is verified to be aligned with all other CEAs in its group by performing surveillance requirement 4.1.3.1.1 at least once per 4 hours.

3. Operation may continue after 7 days provided that Actions 6.c.1, 6.c.2, and 6.c.3 are met.

c. With both CEACS inoperable in any CPC channel, either declare the associated CPC channel(s) inoperable; or, operation may continue provided that:

1. Within 4 hours the DNBR margin required by Specification 3.2.4a (COLSS in service) or 3.2.4b (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

2. Within 4 hours:

a) All CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then perform surveillance requirement 4.1.3.1.1 at least once per 4 hours.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 7 - DELETED

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	SFCP and S/U(1)	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	SFCP	SFCP(2,4),SFCP (3,4), SFCP (4)	SFCP	1, 2
3. Logarithmic Power Level - High	SFCP	SFCP(4)	SFCP and S/U(1)	2#, 3, 4, 5
4. Pressurizer Pressure - High	SFCP	SFCP	SFCP	1, 2
5. Pressurizer Pressure - Low	SFCP	SFCP	SFCP	1, 2
6. Containment Pressure - High	SFCP	SFCP	SFCP	1, 2
7. Steam Generator Pressure - Low	SFCP	SFCP	SFCP	1, 2
8. Steam Generator Level - Low	SFCP	SFCP	SFCP	1, 2
9. Core Protection Calculators	SFCP	SFCP(2,4),SFCP(4,5)	None	1, 2
a. Local Power Density - High	SFCP	SFCP(2,4),SFCP(4,5)	None	1, 2
b. DNBR - Low	SFCP	SFCP(7), SFCP(2,4), SFCP(8), SFCP(4,5)	None	1, 2
c. CEA Calculators	SFCP	SFCP	None	1, 2
10. DELETED				
11. DELETED				
12. Reactor Protection System Logic	N.A.	N.A.	SFCP(11) and S/U(1)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	SFCP(10,11), S/U(1)	1, 2, 3*, 4*, 5*
14. DELETED				
15. DELETED				
16. Reactor Coolant Flow - Low	SFCP	SFCP	SFCP	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(3) Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine or verify acceptable values for the shape annealing matrix elements used in the Core Protection Calculators.

(6) DELETED

(7) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow co-efficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty is included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations.

(9) DELETED

(10) In accordance with the Surveillance Frequency Control Program and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage trip function and the shunt trip function.

(11) The CHANNEL FUNCTIONAL TEST shall be scheduled and performed such that the Reactor Trip Breakers (RTBs) are tested at least every 6 weeks to accommodate the appropriate vendor recommended interval for cycling of each RTB

SPECIAL TEST EXCEPTIONS

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of Functional Unit 9c of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of Functional Unit 9c of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of Functional Unit 9c of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detection Monitoring System pursuant to the requirements of Specifications 4.2.1.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of Functional Unit 9c of Table 3.3-1 are suspended.

6.6 NOT USED

6.7 NOT USED

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and Emergency Operating Procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Not used.

e. Not used.

f. Not used.

g. Modification of core protection calculator (CPC) addressable constants.

These procedures shall include provisions to ensure sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of WCAP-16096-P-A, "CPC Protection Algorithm Software Change Procedure," which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

h. Administrative procedures implementing the overtime guidelines of Specification 6.2.2e., including provisions for documentation of deviations.

i. PROCESS CONTROL PROGRAM implementation.

INDUSTRIAL SURVEY OF TOXIC OR HAZARDOUS CHEMICALS REPORT

6.9.1.9 Surveys and analyses of major industries in the vicinity of Waterford 3 which could have significant inventories of toxic chemicals onsite to determine impact on safety shall be performed and submitted to the Commission at least once every 4 years.

6.9.1.10 A survey of major pipelines ($\geq$ 4 inches) within a 2-mile radius of Waterford 3, which contain explosive or flammable materials and may represent a hazard to Waterford 3, including scaled engineering drawings or maps which indicate the pipeline locations, shall be performed and submitted to the Commission at least once every 4 years.

CORE OPERATING LIMITS REPORT COLR

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or any remaining part of a reload cycle for the following:

3.1.1.1	SHUTDOWN MARGIN – ANY CEA WITHDRAWN
3.1.1.2	SHUTDOWN MARGIN – ALL CEAS FULLY INSERTED
3.1.1.3	MODERATOR TEMPERATURE COEFFICIENT
3.1.2.9	BORON DILUTION
3.1.3.1	CEA POSITION
3.1.3.6	REGULATING AND GROUP P CEA INSERTION LIMITS
3.2.1	LINEAR HEAT RATE
3.2.3	AZIMUTHAL POWER TILT – T_q
3.2.4	DNBR MARGIN
3.2.7	AXIAL SHAPE INDEX
3.6.1.5	AIR TEMPERATURE, CONTAINMENT (Linear Heat Rate, 3.2.1)
3.9.1	BORON CONCENTRATION

6.9.1.11.1 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as follows:

1) "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" (WCAP-11596-P-A), "ANC: A Westinghouse Advanced Nodal Computer Code" (WCAP-10965-P-A), and "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery" (WCAP-10965-P-A Addendum 1) (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, 3.2.4.a.2 for DNBR Margin, 3.1.2.9 for Boron Dilution, and 3.9.1 for Boron Concentrations).

2) "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A (Methodology for Specification 3.1.3.6 for Regulating and Group P CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).

CORE OPERATING LIMITS REPORT COLR (Continued)

3) "Modified Statistical Combination of Uncertainties, CEN-356(V)-P-A, Revision 01-P-A (Methodology for Specification 3.2.4.b.1 and 3.2.4.b.2 for DNBR Margin and 3.2.7 for ASI).

4) "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132-P (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

5) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

6) "Technical Manual for the CENTS Code," WCAP-15996-P-A, Rev. 1 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.3 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating and Group P Insertion Limits, and 3.2.4.a.2 for DNBR Margin).

7) "Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

8) "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (may be used as a replacement for the PHOENIX-P lattice code as the methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, 3.2.4.a.2 for DNBR Margin, 3.1.2.9 for Boron Dilution, and 3.9.1 for Boron Concentrations).

9) "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," WCAP-16072-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Tilt, and 3.2.7 for ASI).

10) "CE 16 x 16 Next Generation Fuel Core Reference Report," WCAP-16500-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, 3.2.4.a.2, 3.2.4.b.1 and 3.2.4.b.2 for DNBR Margin, and 3.2.7 for ASI).

11) "Optimized ZIRLO™," WCAP-12610-P-A and CENPD-404-P-A Addendum 1-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

12) "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," WCAP- 16523-P-A (Methodology for Specification 3.2.4.a.2, 3.2.4.b.1 and 3.2.4.b.2 for DNBR Margin).

13) "ABB Critical Heat Flux Correlations for PWR Fuel," CENPD-387-P-A (Methodology for Specification 3.2.4.a.2, 3.2.4.b.1 and 3.2.4.b.2 for DNBR Margin and 3.2.7 for ASI).