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**Paul Wood**
Manager – Regulatory Assurance
Waterford 3

10CFR50.59 (d)(2)
10CFR72.48 (d)(2)

W3F1-2020-0032

April 30, 2020

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

Subject: Report of Facility Changes, Tests and Experiments and Commitment Changes for two year period ending April 28, 2020
Waterford Steam Electric Station, Unit 3
NRC Docket No. 50-382
Renewed Facility Operating License No. NPF-38

Dear Sir or Madam:

Enclosed is the summary report of facility changes, tests and experiments for Waterford 3, which is submitted pursuant to 10CFR50.59 (d)(2) and 10CFR72.48 (d)(2). This report covers the period from April 28, 2018 through April 28, 2020 and includes copies of the 10 CFR 50.59 and 10 CFR 72.48 Evaluations from this period. The summary report of Commitment Changes for the same time period in line with guidance in SECY-00-0045 and NEI 99-04 are included herein.

If you have any questions regarding this report, please contact Paul Wood, Regulatory Assurance Manager at (504) 464-3786.

There are no new commitments contained in this submittal.

Sincerely,

Paul Wood Digitally signed by Paul Wood
Date: 2020.04.30 17:12:44
-05'00'

PW/rrd

Attachments: Waterford 3 Summary of and Attached 10 CFR 50.59 and 10 CFR 72.48 Evaluations
Waterford 3 Summary of Commitment Changes

cc: NRC Region IV, Regional Administrator

NRC Senior Resident Inspector for Waterford 3

U.S. NRC Project Manager for Waterford 3

**Attachment to**

**W3F1-2020-0032**

**Waterford 3 Summary of and Attached 10 CFR 50.59 and 72.48 Evaluations**

| 10 CFR 50.59 Evaluation Number | Initiating Document | Summary |
|---|---|---|
| 18-02 | EC-0000000530-000 | Ultimate Heat Sink water replenishment for tornado event modification installing several valves and connections and a portable diesel driven pump. |
| 14-01 | EC-0000043927-000 | Vital and Instrument Safety Uninterruptible Power Supply Upgrade Project. |
| 18-03 | EC-0000078061-000 | Technical Requirements Manual 3.3.4 turbine valve testing one-time extension. |
| 19-01 | EC-0000081569-000 | Change to Technical Requirements Manual 3.9.6 adding one-time allowance to move fuel assembly LAHE20 by means other than the Refueling machine. |
| 19-02 | EC-0000073060-000 | Cycle 23 Reload Analysis Report changes to Physics Assessment Checklist (PAC) exceptions and revised Control Element assembly (CEA) drop time. |
| 19-03 | EC-0000082583-000 | Additional Time Critical Operator Action (TCOA) to secure AH-2A(C) following various accidents. |

| 10 CFR 72.48 Evaluation Number | Initiating Document | Summary |
|---|---|---|
| 20-01 | EC-0000086397-000 | During closure operations for MPC-32 S/N 561 at Waterford, personnel were unable to complete the plug weld over one of the 1/4-20UNC set screws (Item 20 on DWG 3753) in the vent port cover plate due to helium pressure buildup beneath the cover plate (ref. CR-WF3- 2020-1455). It is proposed to cut out both the vent and port cover plates, re-perform FHD drying and helium backfill operations, and proceed with closure operations using new Alloy X cover plates (see IPR-2849-112-R0) that utilize stainless steel 1/8" NPT threaded plugs in place of the original port cover plates and set screws. |

## I. OVERVIEW / SIGNATURES[2]

**Facility:** Waterford 3 **Evaluation # / Rev. #:** 07-11 / 0

**Proposed Change / Document:** EC530 – Ultimate Heat Sink Water Replenishment for Tornado Event

**Description of Change:**

This modification provides means to supply additional replenishment water to the ultimate heat sink (UHS) for the design basis tornado event. The previous analysis used an incorrect dry cooling tower (DCT) performance curve to determine the degraded DCT heat duty. Under assumed worst case conditions, it will take longer than previously analyzed for the total plant heat load to lower to the point where shut down cooling can be initiated. Additional replenishment to the wet cooling tower (WCT) basin is needed to support extended operation of the emergency feedwater (EFW) system (to perform the decay heat removal function) and the wet cooling tower portion of the UHS (to provide natural draft cooling of the essential chiller loads).

Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants" states, in part, *"The capacity of the sink should be sufficient to provide cooling both for the period of time needed to evaluate the situation and for the period of time needed to take corrective action. A period of 30 days is considered to be adequate for these purposes."* The water inventory available from the condensate storage pool is currently not adequate to meet this 30 day requirement in the event of tornado damage to the dry cooling towers.

However, the Regulatory Position presented in Section C.1.c further states, *"A cooling capacity of less than 30 days may be acceptable if it can be demonstrated that replenishment or use of an alternate water supply can be effected to assure the continuous capability of the sink to perform its safety functions, taking into account the availability of replenishment equipment and limitations that may be imposed on "freedom of movement" following an accident or the occurrence of severe natural phenomena."* Acceptance testing will be performed to demonstrate the "freedom of movement" capability of the system. Periodic surveillance testing will be established for continued demonstration of pump capability and inventory of necessary equipment. Contracts and/or agreements will be established and in place to ensure a suitable backup pump is available to rent or from other Entergy South sites. Post Return to Service Actions are set up for the Engineering Change to ensure that procedure changes, contracts and/or agreements, and periodic testing are established.

This modification installs hose connections to various non-safety related on-site water sources to be used as alternate replenishment water supplies to the ultimate heat sink (UHS) using either a portable diesel driven pump or the systems' pressure. These on-site water sources include the Fire Protection (FP) system, the Potable Water (PW) system, the Condensate Makeup and Transfer (CMU) system, the Demineralized Water (DW) system, and the Treated Water (TW) system. If these alternate sources of replenishment are rendered unavailable by the tornado event, provision is made for using portable equipment to pump water from the Mississippi River into the Circulating Water inlet line 7CW132-2 to replenish the currently credited non-safety related source of replenishment water. The stagnant raw river water in line 7CW132-2 is currently credited in FSAR 9.2.5.3.2 and 9.2.5.3.3 to replenish the wet cooling tower basins by gravity draining through the existing cross connect to circulating water line 7CW16-31.

The effects of this modification on the water supplies were screened out in the PADs. The scope of this evaluation is to determine the license basis impact of the modification on the auxiliary component cooling water (ACCW) portion of the UHS and the EFW system. Specifically, this evaluation will address the impact of crediting extended operation of the EFW system and the ACCW portion of the UHS to perform functions following a tornado that were previously performed by the component cooling water (CCW) system in conjunction with the DCT portion of the UHS. The evaluation will also address the implementation of additional operator actions to provide the additional UHS inventory replenishment.

[2] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

**Is the validity of this Evaluation dependent on any other change?** ☐ Yes ☒ No

**If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.**

**Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval?** ☐ Yes ☒ No

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**Preparer:** Dale Gallodoro / See EC#530 for Electronic Signatures
Name (print) / Signature / Company / Department / Date

**Reviewer:** John Russo / See EC#530 for Electronic Signatures
Name (print) / Signature / Company / Department / Date

**OSRC:** Kimberly Cook / See EC#530 for Electronic Signatures
Chairman's Name (print) / Signature / Date

07-11
OSRC Meeting #

## II. 50.59 EVALUATION

**Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.** ☐ Yes ☒ No

### Does the proposed Change:

1. **Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?** ☐ Yes ☒ No

   BASIS: Chapters 6 and 15 of the UFSAR were reviewed to identify which accidents previously evaluated in the UFSAR could be initiated or caused by the proposed change. The review confirmed that failure of CCW, ACCW, EFW or UHS components is not an event initiator, nor a contributor to any event initiation for any accident scenario evaluated in the UFSAR.

**2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?**

☐ Yes
☒ No

BASIS:

<u>CCW and ACCW</u>

The failure modes and effects analysis for the ACCW system is described in FSAR Table 9.2-4. The modification has no impact on the failure modes analyzed for the CCW and ACCW systems, since no changes are being made that affect the diesel generators, pump control, or valve failures. The likelihood of occurrence of postulated failures is dispositioned based on the existence of redundant trains of CCW and ACCW. Since the makeup water can be supplied to the separate WCT basins, and the modification does not eliminate the independence and redundancy of existing components, the modification has no impact on the failure modes for the UHS.

<u>EFW</u>

The failure modes and effects analysis for the EFW is described in FSAR Table 10.4-14. The modification does not impact likelihood of a malfunction of either the ac or dc power available to the EFW pumps, since the modification does not interface with the emergency diesel generators or the dc control buses. The modification will not increase the likelihood of a failure of an EFW valve to operate, since the modification does not alter the controls for, or mode of operation of, any EFW valves.

There is no time limit on operation of the EFW pumps listed in the FSAR. The service factor for the EFW pumps is 1.0, and a service factor of 1.0 to 1.08 may be used for an unlimited time if the ambient temperature stays below 40°C (Ref. W3-DBD-003, Rev. 2-8). Additionally, the materials of construction have the resistance, without material deterioration, to withstand a total integrated radiation dose of $1 \times 10^7$ rads from ambient after a period of 40 years of normal operation. The pumps and turbine driver were procured as ASME III Class 3 equipment (Ref. Spec. LOU-1564.117). Based on these considerations, there is sufficient evidence to conclude that the EFW pumps would reliably perform their safety functions during the entire period of operation until such time that SDC could be initiated.

**3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?**

☐ Yes
☒ No

BASIS:
The important-to-safety SSCs affected by the proposed modification include the EFW, CCW and ACCW systems. No modifications are being made to system hardware. However, credit is being taken for extended operation of the EFW system to perform the decay heat removal, and the wet cooling tower (natural draft mode) portion of the UHS to reject essential chiller heat loads following a tornado event. The tornado event is analyzed concurrent with a Loss Of Offsite Power. Section 15.2.1.4.5 of the UFSAR states the radiological consequences of a Loss of Normal AC Power event are bounded by the inadvertent opening of an atmospheric dump valve.

UFSAR Section 15.1.2.4.5.1 states that the radiological analysis considers the secondary steaming pathway from the intact Steam Generator and that this is a minor contributor (< than 1%) to total dose. Therefore, while extended EFW operation is expected to increase steaming from the secondary side, the radiological consequences remain bounded by the radiological consequences of the inadvertent opening of an atmospheric dump valve event (UFSAR Section 15.1.2.4.5.1).

Therefore implementing this modification does not result in an increase in the consequences of an accident previously evaluated in the FSAR.

4. **Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?** ☐ **Yes** ☒ **No**

BASIS: The important-to-safety SSCs affected by the proposed modification include the EFW, CCW and ACCW systems. No modifications are being made to system hardware. However, credit is being taken for extended operation of the EFW system to perform the SDC decay heat removal, and the wet cooling tower (natural draft mode) portion of the UHS to reject essential chiller heat loads following a tornado event. Since no physical or functional modifications are being made to these systems, increased reliance on these systems after a tornado event will not result in a change in the consequences of a malfunction previously evaluated in the UFSAR.

5. **Create a possibility for an accident of a different type than any previously evaluated in the UFSAR?** ☐ **Yes** ☒ **No**

BASIS: The important-to-safety SSCs affected by the proposed modifications include the EFW, CCW and ACCW systems. No modifications are being made to system hardware. However, credit is being taken for extended operation of the emergency feedwater (EFW) system to perform the SDC decay heat removal, and the wet cooling tower (natural draft mode) portion of the UHS to reject essential chiller heat loads following a tornado event. An accident of a different type than previously analyzed in the FSAR would involve the complete loss of emergency feedwater. The depletion of water in both WCT basins without any ability to provide makeup is considered not credible. This modification provides additional replenishment capacity from multiple water sources. Assuming the depletion of all preferably clean water sources stored on site, any additional makeup water could be obtained from the Mississippi River. The ability to supply replenishment water from the river via the circulating water cross tie is part of the existing design and licensing basis as shown on FSAR Figure 10.4-5 Sheet 1.

If any part of the new replenishment system, for example, if the portable pump failed to function, adequate time (approximately 2-3 days) would be available to obtain another pump(s) offsite to perform the same function. Similar or identical pumps and hoses are available from several sources with same day service. Therefore, use of portable equipment to supply water to the UHS does not create the possibility of an accident of a different type than previously evaluated in the FSAR.

| | | | |
|---|---|---|---|
| **6.** | **Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR?** | ☐<br>☒ | **Yes**<br>**No** |

BASIS: The failure modes and effects analyses for the UHS (CCWS and ACCWS Post LOCA) and Emergency Feedwater Systems are presented in FSAR Tables 9.2-4 and 10.4-14, respectively.

<u>CCW</u>

The modification does not create any new failure modes for the installed CCW equipment. This modification merely defers some of the post tornado event cooling functions of the CCW. Specifically, the SDC heat loads and essential chiller heat loads will be rejected by the EFW and ACCW systems until the heat load is within the reduced DCT/CCW capacity.

<u>ACCW</u>

The modification does not create any new failure modes for the installed ACCW equipment. The existing design and licensing basis is based on operating the wet cooling towers in a degraded condition in natural draft mode and on basin inventory replenishment from river water.

Extended operation of the ACCW system using river water does not significantly increase the likelihood of silt blockage of the CCW heat exchangers, or the essential services chillers. The CCW heat exchanger shell side flow rate is monitored in the control room, and a shell side outlet low flow alarm is provided on CP-33. Any silt that may settle in low flow areas inside the heat exchanger can be removed through periodic blowdown through the shell side drain. The ACCW supplied to the chillers flows through the tube side of the condensers where the water velocities are relatively high, so the probability of silt settling inside the tubes is not significant. The water temperature is low (< 105º), so the amount of scale formation in the tubes is not greatly increased over the 30 day period.

Use of a portable makeup pump to supply water to the WCT basins from alternate on-site water supplies only requires placement of the pump discharge hose in the WCT basin without any interconnections to ACCW system components. Since makeup supplied to the WCT is provided through open ended hoses routed to the basins, there is no increased likelihood of overpressurizing ACCW piping or components. All WCT components are designed for the maximum pressure differentials caused by a tornado (FSAR 9.2.5.2), and the WCT below the fans is protected by grating from tornado missiles (W3-DBD-04 3.1.3.5). The ACCW pumps are capable of continuous operation under all operating conditions (W3-DBD-04 3.2.2.1.A.2). The wet cooling towers are designed to operate whenever the heat rejection capacity of the CCW system is exceeded (W3-DBD-04 3.2.2.2) and there is no time limit on the operation of the ACCW in any mode specified in the FSAR. Therefore, operation of the ACCW system for an extended period of 30 days following a tornado in natural draft mode does not increase the likelihood of a malfunction of a SSC.

EFW

The emergency feedwater system failure modes include loss of ac/dc power, failure of a diesel generator to start, failure of an EFW isolation valve to operate, and failure of one EFW pump to start.

This modification makes no changes that would affect the ac/dc power sources or the diesel generator, since the portable pump is engine driven. The modification also does not impact the redundancy of the pumps and valves in the EFW, so there is no impact on the failure of a pump or valve in the EFW.

There is no time limit on operation of the EFW pumps listed in the FSAR. The service factor for the EFW pumps is 1.0, and a service factor of 1.0 to 1.08 may be used for an unlimited time if the ambient temperature stays below 40°C (Ref. W3-DBD-003, Rev. 2-8). Additionally, the materials of construction have the resistance, without material deterioration, to withstand a total integrated radiation dose of $1 \times 10^7$ rads from ambient after a period of 40 years of normal operation. The pumps and turbine driver were procured as ASME III Class 3 equipment (Ref. Spec. LOU-1564.117). Based on these considerations, there is sufficient evidence to conclude that the EFW pumps would reliably perform their safety functions during the entire period of operation until such time that SDC could be initiated.

Potential component malfunctions for use of the portable makeup pumps would be failure of the pump to start. If this occurs, adequate time (2-3 days) exists to obtain another pump(s) from multiple offsite sources, so redundancy and independence also exist for coping with this contingency. Similar or identical pumps and hoses are available from several sources with same day service.

Credit is already taken for use of makeup water from non-safety related sources including the circulating water intake from the Mississippi River. In the existing design, a #8 mesh suction strainer is provided for each EFW pump to prevent the ingestion of any materials that would be detrimental to system operation. Per Specification LOU-1564.084 (Circulating Water Pumps) Table A, Mississippi River water has an average of 231 parts per million (ppm) dissolved solids and 131 ppm suspended solids. For 1,250,000 gallons of makeup, the total solids contained would be 3,762 pounds. Based on past WF3 experience with steam generator cleaning, more than this amount of sludge was removed from the steam generators. Solids that would not remain dissolved or suspended would settle to the top of the tubesheet, but would not clog the downcomer. If downcomer clogging were to occur, then the small amount of feedwater required post-tornado (60-80 gpm per generator) could flow over the top of the chevron sheet and over the tubes and still provide decay heat removal. Therefore, the modification does not create new component failure modes attributable to differences water quality between normal and alternate makeup water sources.

Therefore, this modification does not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the FSAR.

**7. Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?**

☐ Yes
☒ No

BASIS: The use of a portable pump to supply makeup water to the WCT basins does not impact the parameters associated with the fuel cladding, reactor coolant system (RCS) boundary, or containment design pressure.

This modification provides for additional UHS inventory replenishment to supplement the existing raw river water replenishment source in the non-safety related CW piping. The function of the EFW is to provide adequate cooling flow to the secondary plant which in turn cools the RCS such that pressure and temperature are controlled within Technical Specification limits. This function is not affected by the modification, so the parameters affecting stress and pressure in the fuel cladding and the RCS are not affected by the modification.

Use of a portable pump to makeup water to the WCT does not affect the containment boundary since it does alter any containment penetration, does not alter the design of the containment structure, and does not change the pressure inside containment. Therefore the containment fission product barrier is not impacted by the modification.

**8. Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses?**

☐ Yes
☒ No

BASIS: The method of evaluation impacted by this change is the methods of evaluation for other analysis that demonstrate intended design functions will be accomplished, such as analyses that show SSCs will function under design basis natural phenomena. FSAR sections 9.2.5 (UHS) and 10.4.9 (EFW) generally describe the replenishment from the Circulating Water System, but do not describe the calculational or methodical framework used for evaluating EFW and UHS system response following a DBT. As such this change does not result in a departure from the method of evaluation described in the UFSAR.

The FSAR accident analyses assume sufficient quantities of water are readily available for use by the EFW system. Seismic Category I water storage of EFW contains as a minimum, sufficient water to hold the reactor at hot shutdown for two hours, followed by an orderly cooldown until the shutdown cooling system (SDCS) may be initiated. The accidents evaluated in the FSAR are not postulated to occur simultaneously with a tornado.

The use of the portable pump provides a means of supplying makeup water from redundant water sources in the event of tornado damage to the DCT. Since the portable pump is used only to provide makeup upon loss of cooling capacity of the DCT following a tornado, there is no departure from the assumptions used in the safety analyses.

**If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.**

## I. OVERVIEW / SIGNATURES[1]

**Facility:** Waterford 3 **Evaluation # / Rev. #:** # 2014-01 / Rev. #: 1

**Proposed Change / Document:** EC-43927 - Vital and Measurement SUPS Upgrade Project

### Description of Change:

Waterford 3 provides reliable uninterrupted 120 VAC power to vital distribution panels and measurement channel distribution panels using Static Uninterruptible Power Supplies (SUPS). The system is designed to provide reliable, uninterruptible 120 VAC power to the Plant Protection System, Engineered Safety Features (ESF), and other safety-related loads. The redundant SUPS are designed with sufficient separation and isolation so that a single failure will not prevent safe shutdown and cool down of the plant under emergency conditions. The safety related system consists of six SUPS. Division A consists of SUPS A, MA, and MC. Division B consists of SUPS B, MB, and MD.

The proposed activity will install two "Swing" SUPS, SUPS A1 for Division A and SUPS B1 for Division B that can be used to transfer SUPS output power for the Power Distribution Panels (PDPs) from an in-service SUPS to the Swing SUPS. Only one normal SUPS per division (A, MA, MC – Division A; B, MB, MD – Division B) shall be replaced by a swing SUPS in the corresponding Division (Swing A1 – Division A; Swing B1 – Division B). Mechanical interlocks will prevent the swing SUPS from assuming more than one PDP's load at a time. Sync check relays will be used at each transfer panel to verify in phase transfers when swapping PDPs from their normal SUPS to a swing SUPS or vice versa.

The proposed activity will install transfer switch panels called Electric Control Panels (ECPs) to facilitate the electrical transfer or "swing" capability. The ECP will receive power from the normal SUPS and swing SUPS and transfer the power to its associated PDP. Six (6) ECPs will be installed; ECPA, ECPMA, ECPMC, ECPB, ECPMB, and ECPMD.

The proposed activity will also replace older SUPS MA, MB, MC, and MD with new SUPS and their associated PDPs. Replacement SUPS for MA through MD will not have a PDP integral to the enclosure; therefore, new PDP panels will be installed in physically separate locations from that of their associated SUPS. This provides for ease of installation and uniformity.

The proposed activity installs a 4th Static Uninterruptible Power Supply (SUPS) on each train (SUPS A1 and B1) as a backup replacement for the three Train A SUPS (A, MA, MC) and three B SUPS (B, MB, MD) to provide power for the associated Power Distribution Panel (PDP). While the 4th SUPS not only allows for the replacement of any operating SUPS, it also allows for maintenance activities on the SUPS that is not in service to occur. While the 4th SUPS is normally maintained in the OFF condition (de-energized), the design allows the 4th SUPS to be energized provided temperature monitoring is employed, diesel sequencer for the SUPS under test is disabled, and portable battery is used.

The proposed activity will be installed in phases with online and offline activities. Online implementation is expected to start with the installation of: 1) Swing SUPS A1 and B1, 2) new transfer switch panels MA and MD, 3) new power distribution panels MA and MD. Offline installation includes the tie in of: 1) the new PDPs MA and MD, 2) transfer switch panels, 3) Swing SUPS. Later cycle activities will include the online replacement of; 1) SUPS MA and MD, 2) new transfer switch panels MC and MB, 3) new power distribution panels MC and MB. Later offline installation includes the tie in of; 1) new PDPs MC and MB, 2) transfer switch panels. Final cycle activities will include the online replacement of SUPS MC and MD, and installation of new Transfer Switch Panels A and B. PDP A and B cable tie-ins is expected to occur in the following refuel.

This evaluation reviews the addition of swing SUPS and transfer switches while the PAD for EC-43927 evaluates the replacement of the existing SUPS, PDPs, disabling the bypass transformer sequencing function for SUPS MA through MD, providing optional re-sequencing function for Swing SUPS, battery loading, diesel loading, heat loading and fire safe shutdown.

Revision 1 – Adds changes for SUPS A and B bypass to maintain existing resequencing capability (additional diesel load) and provides a discussion on Maintenance testing performed on the out of service SUPS online. This 50.59 is updated for accuracy as phase III of the SUPS project changed method of performing maintenance and added bypass transformers for SUPS A and B back the list of re-sequenced components which increased loading on the Emergency Diesel Generators. Addition of FLEX and transition from Appendix R to NFPA 805 is also included in the change.

[1] The printed name, company, department, and date must be included on the form. Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

The proposed activity will utilize the following Engineering Changes to evaluate and implement the modification:

EC-43927 Parent EC

| Online: No LBDCs Required (unless marked otherwise) | Description | Outage: LBDCs Required | Description |
|---|---|---|---|
| EC-43928 | Swing SUPS A1 Install | EC-43930 | MA Tie-in |
| EC-43929 | Swing SUPS B1 Install | EC-43931 | MD Tie-in |
| *EC-43934 | Replace MA | EC-43935 | MC Tie-in |
| EC-43932 | Replace MD | EC-43936 | MB Tie-in |
| EC-43938 | Replace MC | EC-43939 | A Tie-In |
| EC-43937 | Replace MB | EC-43940 | B Tie-in |

*LBDC Required

**Summary of Evaluation:**
The proposed activity will provide an enhancement by installing two "Swing" SUPS with related support components in each division that can be used to transfer power using Power Distribution Panels (PDPs) from an in-service SUPS to the Swing SUPS. Mechanical interlocks will prevent the Swing SUPS from assuming more than one PDP at a time. The interlocks will also keep multiple Plant Protection System power feeds from being paralleled. Plant analysis has been updated to show additional load on diesel generator, HVAC, MCC and battery loading. While the 4th SUPS is normally maintained de-energized, the analysis concludes it is acceptable to energize the 4th SUPS in each train for maintenance evolutions provided the 4th SUPS is continuously manned, diesel sequencing is disabled, portable battery is used, and temperature monitoring is employed. Protection and coordination calculations have been updated with new breakers. Voltage drop analysis has been performed for the new loads and added cables in the revised circuits. The evaluation shows that proposed change does not require prior NRC approval to implement.

**Is the validity of this Evaluation dependent on any other change?** ☐ Yes ☒ No

**If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.**

**Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval?** ☐ Yes ☒ No

**Preparer[2]:** David Anders / [signature] / DP Engineering / 5-7-18
Name (print) / Signature / Company / Department / Date

**Reviewer[2]:** RICKY TRAN / [signature] / ENS / DE-E / 5/7/18
Name (print) / Signature / Company / Department / Date

**Independent Review[3]:** N/A
Name (print) / Signature / Company / Department / Date

**OSRC:** RAN GILMORE / R.B.Gil / 5-7-18
Chairman's Name (print) / Signature / Date [GGNS P-33633, P-34230, & P-34420; W3 P-151]

18-08
OSRC Meeting #

DP Internal Reviewer
[signature] 5-7-18

2 Either the Preparer or Reviewer will be a current Entergy employee.
3 If required by Section 5.1[3].

R. G. FINKENAUR, III

**II.** **50.59 EVALUATION** [10 CFR 50.59(c)(2)]

**LICENSING BASIS:**

**UFSAR Sections**

7.3 ENGINEERED SAFETY FEATURES SYSTEMS
7.3.1.1 System Description
7.3.1.1.1.6 Redundancy

g) AC power for the actuation system is provided from four separate buses. Power for control and operation of redundant actuated components comes from separate buses. Power source for each bus is from a Static Uninterruptible Power Supply (SUPS). Loss of preferred offsite power does not interrupt power to these vital buses, as described in Subsection 8.3.1.1.1.c.

8.1.4 DESIGN BASIS

8.1.4.1 Offsite Power System

Appendix 8.1A provides the results of a Station Blackout (SBO) Evaluation performed for Waterford 3 in accordance with the requirements of 10CFR50.63. The evaluation demonstrates that equipment will be functional such that Waterford 3 can safely cope with an SBO for four hours.

8.1.4.3 Criteria, Codes and Standards

b) NRC Regulatory Guides:
1) 1.6, Independence Between Redundant (Onsite) Power Sources and Between Their Distribution Systems (3/10/71)
6) 1.32, Use of IEEE Std 308-1971, Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations (8/11/72)
13) 1.75*, Physical Independence of Electric Systems (1/75)
* Indicates that Waterford 3 has taken exception to or interprets the Regulatory Guide. These alternate ways of meeting the intent of the Regulatory Guide are discussed in Subsection 8.3.1.2.

c) Institute of Electrical and Electronics Engineers (IEEE) Standards:
1) IEEE Standard 279-1971, Protection Systems for Nuclear Power Generating Stations, Criteria for Nuclear Power Generating Stations
2) IEEE Standard 308-1971, Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations
7) IEEE Standard 384-1974, Criteria for Separation of Class 1 E Equipment and Circuits

Appendix 8.1A STATION BLACKOUT EVALUATION

LP&L performed an evaluation, ECE89-016, for Waterford 3 for a Station Blackout (SBO) in accordance with 10CFR50.63 using the guidance in NUMARC 87-00 and Regulatory Guide 1.155. There were no hardware changes required for Waterford 3 to cope with an SBO for four hours. Procedural changes are implemented to enhance the ability of Waterford 3 to cope with an SBO. The plant specific evaluation for Waterford 3 demonstrates that equipment will be functional such that Waterford 3 can safely cope with an SBO for four hours.

The Nuclear Utility Group on Station Blackout (NUGSBO), Nuclear Utility Management Resource Council (NUMARC), the NRC, and various technical consulting firms endeavored for several years to resolve the technical issues for an SBO. The resolution addressed the margins of safety, potential malfunctions and accident types, probabilities of malfunctions and accidents, and consequences. The resolution for SBO was established and documented in 10CFR50.63, NUMARC 87-00, and Regulatory Guide 1.155. The plant specific evaluation for Waterford 3 was performed in accordance with the foregoing documents.

The Waterford 3 evaluation was independently verified by Entergy technical personnel and reviewed and approved by cognizant personnel.

The SBO industry resolution, Waterford 3 plant specific evaluation, and independent review and approval of the effort provide additional defense in depth that Waterford 3 will be able to cope with an SBO and that an unreviewed safety question does not exist.

B) SBO Procedure Description

Plant procedures have been reviewed and modified to meet the guidelines in NUMARC 87-00, Section 4, in the following areas:

1. AC power restoration per NUMARC 87-00, Section 4.2.2; LP&L Emergency Procedures for Restoration of Offsite Power to Waterford 3.
2. Severe weather per NUMARC 87-00, Section 4.2.3; OP-901-521 - Severe Weather and Flooding.

Plant procedures have been reviewed and changes necessary to meet NUMARC 87-00 implemented in accordance with 10CFR50.63 in the following area:

1. Station blackout response per NUMARC 87-00, Section 4.2.1; OP-902-005 – Degraded Electrical Distribution Recovery Procedure.

C) Proposed Modifications and Schedule

2. Class 1E Battery(ies) Capacity (Section 7.2.2)

   A battery capacity calculation verified that the Class 1E batteries have sufficient capacity to meet station blackout for four hours.

4. Effects of Loss of Ventilation (Section 7.2.4)

   The assumption in NUMARC 87-00, Section 2.7.1, that the control room will not exceed 120°F during a station blackout has been assessed.

   The control room at Waterford 3 does not exceed 120°F during station blackout. Therefore, the control room is not a dominant area of concern.

   Reasonable assurance of the operability of station blackout equipment in the areas containing potential heat sources have been assessed using Appendix F to NUMARC 87-00 or the Topical Report. No modifications or associated procedures are required to provide reasonable assurance for equipment operability.

8.3.1 AC POWER SYSTEMS

8.3.1.1.1 General

c) 120 Volt Uninterruptible (Vital) AC System

A 120V uninterruptible ac system has been provided to supply the Plant Protection System control and instrumentation channels. The 120V uninterruptible AC system consists of rectifier/inverters and power distribution panels. Each inverter is normally supplied through its rectifier from a 480V ESF MCC. Should this supply fail, the inverter is supplied automatically from a 125V ESF battery.

The Plant Protection System (PPS) uses four inverters, two from each division, to supply the four measurement channels.

The other safety-related control and instrumentation systems are connected to two inverters, one for each Division A and B. A seventh inverter without battery back-up, and eighth inverter with its own battery, are used to supply other important but nonsafety-related loads. The plant monitoring computer is supplied from a ninth inverter, with its own battery.

The four PPS ac systems and two ac safety-related control and instrumentation systems are ungrounded while the remaining ac systems have solidly grounded neutrals.

Each system is arranged so that any type of single failure or fault will not prevent proper protective action of the safety related systems.

Power and control cables for the 120V uninterruptible ac systems are rated 600 V 90°C with ethylene-propylene rubber or cross-linked polyethylene insulation, flame-resistant jacket and copper conductors of the cables are sized to carry the maximum available short circuit current for the time required by the circuit breaker or fuse to clear the fault. These cables are normally sized for continuous operation at 125 percent of nameplate full-load current. (NOTE: Due to cable tray fill and fire/separation wrap requirements some cables have been derated. Engineering calculations demonstrate the ampacity of these cables are properly sized for the connected loads.)

8.3.1.1.2.4 Manual and Automatic Interconnections Between Buses, Between Buses and Loads, and Between Buses and Supplies.

There are no connections, either manual or automatic, between buses of different divisions. There are also no interconnections between the 120V uninterrupted ac (nuclear instrumentation) buses, although the two supply inverters for channels A and C are driven normally by 480V feeders from separate Division A MCCs.

(Emergency dc supply to these two inverters is also by separate feeders from the Division A Battery 3AS). Similarly, inverters B and D are powered by separate feeders from Division B supplies.

Loss of the ac feeder to any inverter results in automatic assumption of load by the DC feeder because the ac input is rectified and the resultant dc output is "auctioneered" with the DC feeder input. Thus the supply with the higher voltage (normally the ac feeder) supplies the inverter.

8.3.1.1.2.6 Redundant Bus Separation

Separation of redundant 4.16 kV and 480V redundant power centers, the 480V redundant MCCs and power panels, the 120V uninterruptible ac buses and inverters and the 125V DC batteries, chargers and distribution panels has been accomplished through spatial separation or provision of fire resistant barriers. The two redundant diesel generators are housed in separate fire resistant rooms in Reactor Auxiliary Building which is a seismic Category I structure.

8.3.1.1.2.10 Instrumentation and Control Systems with Assigned Power Supply

The Plant Protection System (PPS), including the Reactor Protection Systems (RPS) and core protection calculators and other instrumentation and control systems provided for monitoring and controlling the reactivity, temperature and other vital parameters within the reactor, is supplied with power from the four uninterruptible AC inverters described in Subsection 8.3.1.1.1 (c). There are four separate channels in these control systems, each of which operates at 120VAC ungrounded, from one of the four buses 3MAS, 3MB-S, 3MC-S and 3MD-S. Buses 3MA-S and 3MC-S receive power from inverters supplied from Division A power and buses 3MB-S and 3MD-S receive power from inverters in Division B. Thus, independence of the four channels from each other extends back to either the 480V safety-related power center buses 3A31-S and 3B31-S, or the 125VDC distribution panels 3A-DC-S/3A1-DC-S and 3B-DC-S/3B 1-DC-S.

The other safety-related control and instrumentation systems receive power from two inverters similar to those of the PPS, and also described in Subsection 8.3.1.1.1 (c).

Each inverter is supplied from a safety-related MCC, with automatic transfer to battery supply on ac failure. Since the AC and DC supplies for the two inverters are taken from the same Division (A or B) as the inverter serves, full separation between divisions is assured.

Controlled actuators or final devices, such as motor operated valves, receive power from safety-related MCCS, if AC, and from the 125 V batteries, if DC; larger devices, such as pumps, are powered from 480V power centers or 4160V switchgear, and control power is supplied in these cases from the 125V battery of the appropriate division.

8.3.1.2.13 Regulatory Guide 1.75-1975

The Class 1E portions of the Onsite Electric System comply with the positions of this guide, as follows:

c) Position C3. As far as possible, redundant equipment is located in separate compartments within a seismic Category I structure. Where this is not possible, barriers or physical separations are used as described in Subsection 8.3.1.2.19.

8.3.1.2.14 IEEE Standard 279-1971

The provisions of this standard relate mainly to the Reactor Protective System and are discussed in Chapter 7. The electrical system supplying power to the Reactor Protective System has been designed to ensure that failures in the supply system have no worse consequences than failures in the Reactor Protective System, as follows:

a) Power supply to the protection systems is from four (one for each channel) power supply inverters as described in Subsection 8.3.1.1.1(c). No random single failure in any one inverter will degrade the performance of the other three. With one measurement channel bypassed for testing, failure of a second channel inverter will still leave two channels functional, thus providing protection without unnecessary tripping (because of the "two out of four" logic).
b) Any one of the four power supply units can be isolated for maintenance at the same time as the remaining protective channel equipment is being maintained.
c) Action of the manual transfer of the 120V bus to the bypass transformer is annunciated in the main control room.
d) Each power supply unit is so constructed as to facilitate repair by replacement of defective components or modules, to ensure a minimum of downtime.

8.3.1.2.15 IEEE Standard 308-1971

b) AC Power Systems
   1) Alternating current power systems include power supplies, a distribution system and load groups arranged to provide ac electric power to the Class 1E loads. Sufficient physical separation, electrical isolation and redundancy have been provided to prevent the occurrence of common failure modes in the Class 1E systems.
   2) The electric loads have been separated into two redundant groups.
   3) The safety actions by each group of loads are redundant and independent of the safety actions provided by the redundant counterparts.
c) Distribution System
   2) Physical isolation between redundant counterparts ensures independence.

8.3.1.2.19 IEEE Standard 384-1974

a) General Separation Criteria

   Equipment and circuits requiring separation have been identified on drawings and in the field in a distinctive manner as described in Subsection 8.3.1.3.

   All control and power equipment and cables of systems in each safety related division have been separated from those of the other division and from those of non-safety related systems, except as noted in Subsection 8.3.1.2.13.

   Class 1E equipment is installed in safety class structures and where equipment of both divisions is contained in a single room, separation is provided by incombustible barriers.

g) Specific Separation Criteria - Control Boards

   With the exceptions of the two diesel-generator local control boards (paragraph (c) above), all safety related control boards are located in the main control room. The main control room is free from high pressure steam or water piping and from major rotating machinery, and control boards are not exposed to pipe whip, jet impingement or missiles.

   Redundant Class 1E equipment is mounted on separate panels wherever possible. Where separate panels are not feasible, instrumentation and other equipment is grouped so that the minimum distance between items of different safety divisions or measurement channels is six in., where this clearance is not possible, a steel barrier is used.

   Wiring of each safety division or measurement channel is bundled and identified (see Table 8.3-12); where wiring of one division or channel must traverse an area dedicated to another division or channel, steel conduit or solid tray with cover is used.

8.3.1.4 Independence of Redundant Systems

The redundant systems are designed to be physically independent of each other so that failure of any part or the whole of one train, channel or division will not prevent safe shutdown of the plant.

The Class 1E electric systems are designed to ensure that the design basis events listed in IEEE 308- 1971 will not prevent operation of the minimum amount of ESF equipment required to safely shutdown the reactor and to maintain a safe shutdown condition.

The Class 1E power system is designed to meet the requirements of IEEE 279-1971, IEEE 308-1971, 10CFR50, including Appendices A and B, and Regulatory Guide 1.6. ESF loads are separated into two completely redundant load groups. Each load group has adequate capacity to start and operate a sufficient number of ESF loads to safely shutdown the plant, without exceeding fuel design limits or reactor coolant pressure boundary limits, during normal operation or design basis event. As required by IEEE 308 and 10CFR50 (General Design Criterion 17) each redundant ESF load can be powered by both onsite and offsite power supplies. Two diesel generators, one on each ESF bus, will furnish the required emergency

8.3.2.2.1.5 IEEE-308-1971

For the analysis per Principal Design Criteria of IEEE-308-1971, see Subsection 8.3.1.2. The following presents an analysis per supplementary Design Criteria as applicable to the Class 1E DC system.

Dependable power supplies have been provided for the Plant Protection System. Two independent DC and four independent AC power supplies have been provided for control and instrumentation of these systems. The independent DC supplies are provided by distribution circuits from each of two redundant DC distribution panels. Independent AC supplies are provided by the four inverters and associated 120VAC buses. Refer to Subsection 8.3.1.1 for further description of these 120V uninterruptible AC power supplies.

Since each inverter is normally powered from an AC supply with DC backup, the failure of a battery or battery charger will not in any way effect the operation of the required ac loads from the inverter, unless there is a simultaneous failure of the AC feeder.

FSAR Table 8.3-9 “120V UNINTERRUPTIBLE VITAL AC SYSTEM SINGLE FAILURE ANALYSIS”

| FAILURE | CAUSE | CONSEQUENCES AND COMMENTS |
|---|---|---|
| 1. 120VAC power to buses 3MA-S, 3MB-S, 3MC-S or 3MD-S | a. Bus fault<br>b. Cable fault<br>c. Failure of a distribution breaker to clear a fault | a, b, c,. The result will be the loss of 120 volt uninterruptible AC power supply to one of the four channels of the protection system. As a two out of four criterion is used in all logic circuits, the remaining three channels ensure safe, but not false, shutdown. The 120V uninterruptible AC system has been designed as an ungrounded system. The reliability of any channel is consequently greatly enhanced. |
| 1. 2. Any distribution Feeder | a. Cable fault | a. This will result in the loss of power to the connected Feeder loads. The redundant loads in the remaining three channels are adequate to ensure safety. |
| 2. 3. Loss of 480VAC power to SUPS | a. MCC bus fault<br>b. cable fault | a, b. The SUPS will be supplied by the battery without interruption of output power. |

**General Design Criteria**

The following is General Design Criterion 17 from Appendix A of 10CFR50.

Criterion 17 – "Electric Power Systems" An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

**Regulatory Guides**

Waterford 3 is committed to Regulatory Guide 1.32, 1972 “Use of IEEE STD 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations". This guide endorses IEEE 308 1971 “Class IE Electric Systems for Nuclear Power Generating Stations.” IEEE 308 provides the following:

4. Principal Design Criteria
4.1 General
The Class 1E power systems shall be designed to assure that no design basis event will cause: (1) A loss of electric power to a number of engineered safety features, surveillance devices, or protection system devices sufficient to jeopardize the safety of the station; (2) A loss of electric power to equipment that could result in a reactor power transient capable of causing significant damage to the fuel or to the reactor coolant system.

5. Supplementary Design Criteria
5.1. Class IE Electric Systems
5.1.1. Description. The Class IE electric systems shall consist of an alternating-current power system, a direct current power system, and an instrumentation and control power system. Figure 1 illustrates one possible arrangement of the Class IE electric systems for a single-unit generating station.
5.1.2. Function. The Class IE electric systems shall provide acceptable power to the station during and following any design basis event.

5.2 Alternating-Current Power Systems.
5.2.1. General. The alternating-current power systems shall include power supplies, a distribution system, and load groups arranged to provide alternating-current electric power to the Class 1E loads. Sufficient physical separation, electrical isolation, and redundancy shall be provided to prevent the occurrence of common failure mode in the Class IE systems. Design requirements shall include, but are not necessarily limited to, the following:
1) Redundant Load Groups: The electric loads shall be separated into two or more redundant load groups.
5.2.3 Preferred Power Supply.
(2) Function. The preferred power supply shall furnish electric energy for the shutdown of the station and for the operation of emergency systems and engineered safety features. This does not preclude its use for other functions.
(3) Capability. The preferred power supply shall be capable of starting and operating all required loads.

5.4. Vital Instrumentation and Control Power Systems
5.4.1. General. Dependable power supplies are required for the nuclear generating station's vital instrumentation and control systems, including:
1) The nuclear plant protection, instrumentations, and control systems.
2) The engineered safety features instrumentation and control systems.
5.4.2. Design Requirements. The diverse arrangements, special requirements, and complexity of these systems preclude a detailed delineation of their power supply requirements. However, power must be supplied to these systems in such a manner as to preserve their reliability, independence, and redundancy. Typically, one or more of the following may be required:
3) Two or more independent alternating-current power supplies having a degree of availability, compatible with the system it serves.

Waterford 3 is committed to Regulatory Guide 1.155, 1998 "Station Blackout"
The Station Blackout evaluation is documented in UFSAR Section 8, Appendix 8.1A which states *"LP&L performed an evaluation, EC-E89-016, for Waterford 3 for a Station Blackout (SBO) in accordance with 10CFR50.63 using the guidance in NUMARC 87-00 and Regulatory Guide 1.155."*

- NUMARC 87-00 section 2.2.1 states in the initial plant condition assumptions that:
  *(1) The station blackout event occurs while the reactor is operating at 100% rated thermal power and has been at this power level for at least 100 days.*
  *(2) Immediately prior to the postulated station blackout event, the reactor and supporting systems are within normal operating ranges for pressure, temperature, and water level.* ***All plant equipment is either normally operating or available from the standby state.***
- RG1.155 section C states *"This regulatory guide describes a means acceptable to the NRC staff for meeting the requirements of § 50.63 of 10 CFR Part 50. NUMARC-8700 also provides guidance acceptable to the staff for meeting these requirements. Table 1 provides a cross-reference to NUMARC-8700 and notes where the regulatory guide takes precedence."*

**Standard Review Plan**
Standard Review Plan, NUREG-0800; Revision 2, July 1981, 8.3.1 A-C POWER SYSTEMS (ONSITE) provides the following:

6. Vital Supporting Systems

The PSB will review those auxiliary systems identified as being vital to the operation of safety-related loads and systems. The PSB reviews the instrumentation, control, and electrical aspects of the vital supporting systems to ensure that their design conforms to the same criteria as those for the systems that they support. Hence, the review procedure to be followed for ascertaining the adequacy of the vital supporting systems is the same as that discussed herein for the onsite systems. In essence, the reviewer first becomes familiar with the purpose and operation of each vital supporting system, including its components arrangement as depicted on functional P&IDs. Subsequently, the design criteria, analyses, and description and implementation of the instrumentation, control and electrical equipment, as depicted on electrical drawings, are reviewed to verify that the design is consistent with satisfying the acceptance criteria for Class 1E systems. In addition, it is verified that the vital supporting system redundant instrumentation, control devices, and loads are examined to verify that they are powered from the same redundant distribution system as the system that they support. The PSB will also verify that the vital supporting systems which are associated with the emergency diesel engine such as the fuel oil storage and transfer system, cooling water system, starting air system and lubrication system are in accordance with the acceptance criteria.

**Technical Specifications**

3.8.3.1 "Onsite Power Distribution Systems – Operating," and
3.8.3.2 "Onsite Power Distribution Systems – Shutdown" state the following:

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following Engineered Safety Features (ESF) and Static Uninterruptible Power Supply (SUPS) busses shall be energized in the specified manner. The tie breakers from the Train AB Busses shall be connected to either Train A or Train B.

a. Train A A.C. ESF busses consisting of:
   1. 4160-volt ESF Bus #3A3-S
   2. 480-volt ESF Bus #3A31-S

b. Train B A.C. ESF busses consisting of:
   1. 4160-volt ESF Bus #3B3-S
   2. 480-volt ESF Bus #3B31-S

c. Train AB A.C. ESF busses consisting of:
   1. 4160-volt ESF Bus #3AB3-S
   2. 480-volt ESF Bus #3AB31-S

d. 120-volt A.C. SUPS Bus #3MA-S energized from its associated inverter connected to D.C. Bus #3A-DC-S*.

e. 120-volt A.C. SUPS Bus #3MB-S energized from its associated inverter connected to D.C. Bus #3B-DC-S*.

f. 120-volt A.C. SUPS Bus #3MC-S energized from its associated inverter connected to D.C. Bus #3A-DC-S*.

g. 120-volt A.C. SUPS Bus #3MD-S energized from its associated inverter connected to D.C. Bus #3B-DC-S*.

h. 120-volt A.C. SUPS Bus #3A-S energized from its associated inverter connected to D.C. Bus #3A-DC-S.

i. 120-volt A.C. SUPS Bus #3B-S energized from its associated inverter connected to D.C Bus #3B-DC-S.

j. 125-volt D.C. Bus #3A-DC-S connected to Battery Bank #3A-S.

k. 125-volt D.C. Bus #3B-DC-S connected to Battery Bank #3B-S.

l. 125-volt D.C. Bus #3AB-DC-S connected to Battery Bank #3AB-S.

APPLICABILITY: MODES 1, 2, 3, and 4.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One division of A.C. ESF busses consisting of one 4160 volt and one 480-volt A.C. ESF bus (3A3-S and 3A31-S or 3B3-S and 3B31-S).

b. Two 120-volt A.C. SUPS busses energized from their associated inverters connected to their respective D.C. busses (3MA-S, 3MB-S, 3MC-S, or 3MD-S).

c. One 120-volt A.C. SUPS Bus (3A-S or 3B-S) energized from its associated inverter connected to its respective D.C. bus.

d. One 125-volt D.C. bus (3A-DC-S or 3B-DC-S) connected to its associated battery bank.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, or load movements with or over irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible. |

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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The failure modes of the swing SUPS are the same as the normal SUPS. The SUPS failures are given in FSAR Table 8.3-9 “120V UNINTERRUPTIBLE VITAL AC SYSTEM SINGLE FAILURE ANALYSIS.” The new mechanical transfer switches include new, mostly passive components that could fail. Failure of the new components has the same resultant consequences as those listed in Table 8.3-9.

**Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.** ☐ Yes ☒ No

**Does the proposed Change**:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

Loss of a SUPS is not the initiating event for any accident scenario described in Chapter 15 of the USAR. The addition of a Swing SUPS per division and the associated transfer switches adds the capability to take a SUPS out of service due to a failure or for maintenance without deenergizing the associated loads. While additional components are being added to the power supply system, they are of the same quality as the existing equipment and introduce no new failure mode. Therefore, the frequency of occurrence of such an accident previously evaluated in the UFSAR is unaffected by this modification.

The transfer of a normal SUPS powering a PDP to the swing SUPS requires a manual operation. The transfer switches isolate the output from swing SUPS to the various PDPs via interlocks, barriers and breakers. The new SUPS inverters have the same form, inherent function and the same quality classification as the existing SUPS inverters.

During normal plant operation, the SUPS provides 120 VAC power from the rectifier through the inverter. On loss of AC power, the battery assumes the load. If the sources supplying the inverter or the inverter itself fails to produce the required 120 VAC, the bypass feed can assume the load of the PDP. On SUPS A and B, transfer of the PDP from inverter to bypass feed is automatic via the static switch. The static switch is an electronic switch which is used to switch the supply feed from inverter output to bypass output on loss of inverter. The existing SUPS MA through MD have bypass feeds but the transfer is manual; therefore, if a failure occurs which results in loss of 120VAC from the inverter, a panel outage will occur. Adding an automatic bypass feed function to SUPS MA through MD is an enhancement which will prevent the loss of PDP loads on loss of inverter output. Failure of a new static switch would cause the loss of the PDP loads: however, this condition is bounded by the existing analysis. Also, the single failure criteria is still maintained as given in FSAR Table 8.3-9 "120V Uninterruptible Vital AC System Single Failure Analysis" and will be updated to show the transfer switch.

The Swing SUPS will have the same ratings as the existing SUPS for power but will have a lower DC voltage operating capability. The Swing SUPS will have a bypass transformer and an automatic transfer switch that will transfer load from the SUPS to the bypass feed should the SUPS inverter fail like the other SUPS have after all phases of the EC are installed.

The bypass sources for SUPS MA through MD require a larger 480VAC MCC bucket and require re-arrangement to accommodate the larger size. SUPS A and B bypass source sizes remain unchanged. Moving the SUPS MA through MD bypass sources is acceptable as these will no longer be re-sequenced onto the bus during a loss of power event. Disabling the re-sequencing for the bypass supplies on SUPS MA through MD is acceptable because the SUPS bypass supplies perform no safety function and are not credited for the mitigation of any accident described in the FSAR. Disabling the bypass re-sequencing was necessitated by the limited capacity of Diesel fuel oil. By disabling the bypass re-sequencing circuits, Diesel fuel oil consumption was reduced which adds margin for Diesel fuel oil capacity. Again, the single failure criteria is still maintained as given in FSAR Table 8.3-9 "120V Uninterruptible Vital AC System Single Failure Analysis".

Swing SUPS A1 and B1 bypass sources will be modified with a selector switch that allows either manual or auto (resequencing) to occur based on which SUPS the Swing SUPS is being used to functionally replace. When the selector switch for the Swing SUPS bypass is placed in Manual, during normal operations (startup, steady state operations, and shutdown of the SUPS) and under

Loss of Offsite Power events, the bypass breakers for these SUPS will not re-sequence and will operate as manual breakers requiring Operator Action for re-closure locally at the MCC for the given supply. When the selector switch for the Swing SUPS bypass is placed in Auto, during normal operations (startup, steady state operations) and under Loss of Offsite Power events, the bypass breakers for these SUPS will automatically re-sequence and will operate as automatic breakers requiring no Operator Action for re-closure locally at the MCC for the given supply. The Auto position will be selected when the Swing SUPS are used in place of SUPS A and B which have auto resequencing bypass transformers thereby maintaining the design function to auto re-sequence under restoration of loss of power conditions.

Since there are two Swing SUPS being installed, one per division; the divisional Swing SUPS will tie only to the PDPs for that Division and to only one PDP in a Division at a time. This will maintain the divisional separation. Also required is channel separation between the Plant Protection System (PPS) power sources. As stated above, the transfer switches will contain interlocks, barriers and breakers which will ensure that channel separation is maintained on the secondary of the swing SUPS. The Swing SUPS will assume the function of the channel/division SUPS it replaces.

Separation will be maintained as described in FSAR sections 8.3.1.1.1(c), 8.3.1.1.2.4, 8.3.1.1.2.6, 8.3.1.1.2.10, 8.3.1.1.2.13, 8.3.1.1.2.14, 8.3.1.1.2.15 and 8.3.1.1.2.19 (which include compliance with Reg. Guide 1.75 and IEEE 384 for separation requirements) as describe above.

Calculation ECE90-006 "Emergency Diesel Generator Loading and Fuel Oil Consumption" is revised by each implementing phase to show the changes in loading. The new SUPS MA through MD present a negligible increase loading on the diesel generators. For online testing of the out-of-service SUPS, the re-sequencing circuit for the out-of-service SUPS will be disconnected for the duration of testing performed online. Disabling of the resequencing circuit will prevent the out-of-service SUPS from presenting itself as load to the Emergency Diesel Generators (EDGs) during loss of power events. Therefore the replacements of SUPS MA through MD, the addition of SUPS A1 and B1, and the allowance for online testing of the out of service SUPS does not adversely affect the performance or design basis function of the Emergency Diesel Generators.

The related HVAC calculations at WF3 are 5-D (design temperature requirements) and 5-T (required flow / chiller capacity). Calculations, ECE89-003 and ECE89-005, document switchgear room heat loads at the Inception of an SBO for Switchgear Rooms A and B and evaluate the heat loading of replacement SUPS with the fourth SUPS de-energized. Calculation ECE96-001, "Heat Released by Electrical Equipment in the RAB SWGR Area" is also updated by the change. ECS14-004, "WF3 FLEX Switchgear And DC Equipment Rooms BDBEE Heat-Up Analysis" is updated by the change to show conclusions in the calculation remain valid for beyond design basis event impacts on room temperature during an extended loss of offsite power. The proposed changes, while increasing heat load, remain within the capability of the ventilation system and chiller capacity. Therefore, the proposed changes are not adverse as both the ventilation system and chiller continue to perform their design function. Calculation 5-D delineates the required temperatures that must be maintained for equipment operability under normal and accident conditions. Calculation 5-D temperatures must not be exceeded or equipment in the designated area is not operable. Maintenance testing of the out of service SUPS requires temperature monitoring and test termination if temperatures approaches limits established in EC-43927. Terminating the test prior to exceeding the temperature limits established by EC-43927 will maintain the design basis temperatures assumed in calculations 5-D and 5-T.

Calculation ECE89-018 "Control Circuit (120 VAC and 125 VDC) Maximum Loop Length DV" is revised to include new cable lengths and load. No issues voltage drop issues were identified.

Calculations ECE91-058 "3A-S A Train Calculation for Station Blackout", ECE91-059 "3B-S B Train Calculation for Station Blackout", ECE91-061 "Battery 3A-S Cell Sizing", ECE91-062 "Battery 3B-S Cell Sizing" have been revised to show the new SUPS inverters. Loading on each battery increased within the available margin on the batter. While battery margin is reduced, there is still ample margin available for future load growth requirements and the batteries remain capable of performing their design function. ECE14-004, "Battery 3A-S "A" Train Calculation for Flex Event" and ECE14-005, "Battery 3B-S "B" Train Calculation for Flex Event" were updated to show the additional loading on the batteries. The 12.5 hour requirement (30 minutes of margin) for at least one of the two trains to supply DC power is maintained; therefore, the FLEX Phase I analysis as described in WF3-SA-14-0002 is not impacted.

Calculation ECE94-005, "Coordination Study of 480 SWGR to 120V Panel Molded Circuit Breakers" is revised to add new Swing SUPS A1-S and B1-S to show coordination is maintained with the new breakers feeding the swing SUPS.

ECF13-001, "NFPA 805 NSCA Calculation" is updated to reflect the new SUPS and swing system. The SUPS modification does not change the Nuclear Safety Performance Methodology of NFPA 805, Chapter 2 or the Radioactive Release Evaluation. ECF09-005, "NFPA-805 Transition Non Power Operating Mode (NPO)" is updated. Based on the EC maintaining the divisional separation between Train A and Train B power supplies, Non Power Operations (NPO) compliance is maintained in all Fire Areas. Based on quantitative assessment performed in WF3-FP-17-00001, the change in CDF and LERF are acceptable.

Loss of a SUPS is not the initiating event for any accident scenario described in Chapter 15 of the USAR. Therefore, the frequency of occurrence of such an accident previously evaluated in the UFSAR is unaffected by this modification.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
The new Swing SUPS and the replacement SUPS are equipped with automatic static switches. While this feature mitigates the effect of a SUPS inverter failure, static switch failure can still cause the loss of SUPS output, a condition which is unchanged from the existing design as stated in the single failure analysis in FSAR Table 8.3-9. Also, the new transfer switches add new components that could fail, but this failure is also bounded by the single failure analysis in the FSAR. All of the new or replacement components are manufactured to the same quality standards and are seismically qualified as the original equipment and therefore have the same likelihood of malfunction or failure. Additionally, the swing SUPS and transfer switches will maintain the necessary isolation and separation from other PPS channels by use of interlocks, barriers and breakers. For these reasons, the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR is not increased.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
The new swing SUPS are designed to be installed replacements for any one of the existing SUPS in the associated electrical division. The failure modes and effects of the new swing SUPS are the same as those for the existing equipment (loss of the associated SUPS bus). Online testing of the out of service SUPS has been evaluated to have no impact on conclusions reached in the capability of systems used to mitigate the consequences of an accident (diesel, HVAC, battery); therefore, the consequences of any accident involving SUPS bus loss or ancillary impacts on other systems due to online testing (diesel, HVAC, battery) are unchanged by this modification. Furthermore, loss of a SUPS is not the initiating event for any accident scenario described in Chapter 15 of the USAR. The dose analysis in FSAR Chapter 15 assumes the most limiting single failure in conjunction with the event. Failures that initiate the event or occur as a consequence of the event are not considered the single failure. The analysis also assumes a loss of off-site power, if it is more limiting. Therefore, the proposed change is bounded by the existing dose analysis and does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
No greater reliance on the SUPS is created by the replacement of existing MA through MD SUPS or addition of the Swing SUPS. The addition of a Swing SUPS per division and the associated transfer switches adds the capability to take a SUPS out of service due to a failure or for maintenance without deenergizing the associated loads. No new failure modes or effects are introduced by this modification. The modification aids in the mitigation of the consequences of a SUPS malfunction; therefore, there is no increase in the consequences in question.
The dose analysis in FSAR Chapter 15 assumes the most limiting single failure in conjunction with the event. Failures that initiate the event or occur as a consequence of the event are not considered the single failure. The analysis also assumes a loss of off-site power, if it is more limiting. Therefore, the proposed change is bounded by the existing dose analysis and does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
The replacement SUPS have the same function as the existing equipment and introduce no new failure modes. The Swing SUPS and transfer switch provide the ability to replace any normal SUPS within its division. Since the Swing SUPS is the same as the normal SUPS, it will allow continued operation of the PPS channel or other safety related instrumentation and control functions. The use of isolation devices, barriers and breakers in the transfer switches will not allow the paralleling with the normal SUPS or the tying of two different PPS channel power supplies together. Therefore, the possibility for an accident of a different type than any previously evaluated in the UFSAR is not created by this modification.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
The replacement SUPS have the same function as the existing equipment and introduce no new failure modes. The failure of a Swing SUPS or transfer switch has the same results as the normal SUPS which is the loss of power to the respective PPS. This causes the loss of one protective channel from the PPS; however, this condition is analyzed as stated in FSAR section 8.3.1.2.14 which reads as follows:

> "The provisions of this standard (IEEE 279) relate mainly to the Reactor Protective System and are discussed in Chapter 7. The electrical system supplying power to the Reactor Protective System has been designed to ensure that failures in the supply system have no worse consequences than failures in the Reactor Protective System, as follows:
>
> a) Power supply to the protection systems is from four (one for each channel) power supply inverters as described in Subsection 8.3.1.1.1(c). No random single failure in any one inverter will degrade the performance of the other three. With one measurement channel bypassed for testing, failure of a second channel inverter will still leave two channels functional, thus providing protection without unnecessary tripping (because of the "two out of four" logic)."

The proposed modifications have no impact on this analysis; therefore; the possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR is not created.

7. Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered? ☐ Yes ☒ No

   BASIS:
   The replacement SUPS have the same function as the existing equipment and introduce no new failure modes. The Swing SUPS has the same electrical properties as the normal SUPS; therefore, the Swing SUPS will provide power to the PPS equipment as the normal SUPS. This allows the PPS to continue to provide a protective function which does not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

   BASIS:
   The safety related batteries, 3A-S and 3B-S, are still sized in accordance with FSAR section 8.3.2.1.1 "Batteries" and the SUPS loading remains unchanged. All other updates supporting this change are completed using existing methodologies; therefore, this change does not result in a departure from a method of evaluation described in the FSAR used in the design bases or safety analyses.

**If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.**

## I. OVERVIEW / SIGNATURES[1]

**Facility:** Waterford-3 **Evaluation # / Rev. #:** 19-08 / 0

**Proposed Change / Document:** EC73060 Waterford-3 Cycle 23 Reload

**Description of Change:**
Engineering change EC73060 documents the evaluation of the design and performance of the Waterford-3 Cycle 23 reload core and the output documents from the reload process. These changes include the Reload Analysis Report (RAR) [WF3-NE-18-00001], Core Operating Limits Report (COLR), and Updated Final Safety Analysis Report (UFSAR). The reload analyses considered changes relative to Cycle 22 which included cycle specific reload core characteristics (i.e., cycle length, loading pattern, power distributions, etc.) and changes in the unit itself that could impact reload analyses. A Licensing Basis Document Change Request (LBDCR) 18-026 has been initiated for the COLR changes and LBDCRs 18-027 and 18-028 have been initiated for the UFSAR changes.

The Cycle 23 Reload Analysis Report was prepared to document changes in nuclear, thermal-hydraulic, and mechanical design of the reactor core. As such, the reload report provides the bases for the operation of the Cycle 23 fuel design. All analyses and assessments were performed using NRC approved methodologies. No Technical Specification changes were required to implement the Cycle 23 reload.

The Cycle 23 reload core will continue operation with a complete core of the Next Generation Fuel (NGF) Design. There are no plant changes relative to Cycle 22 which impacted the RAR. There are no Reactor Coolant System (RCS) chemistry changes for Cycle 23. Zinc addition will continue for Cycle 23 at a rate similar to that of Cycle 22.

EC73060 Cycle 23 Reload Process Applicability Determination (PAD) identified the following adverse changes.

- Physics Assessment Checklist (PAC) exception for the Loss of Coolant Accident (LOCA) Unrodded Pin Census
- PAC exception for the LOCA Hot Rod Minimum Pin-to-Box Ratio
- PAC exception for the Radial Power Falloff
- Revised control element assembly (CEA) drop time

This 50.59 evaluation addresses the adverse changes identified in the PAD.

[1] The printed name, company, department, and date must be included on the form. Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

**Summary of Evaluation:**
Engineering change EC73060 documents the evaluation of the design and performance of the Waterford 3 Cycle 23 reload core and the output documents from the reload process. These changes include the RAR, COLR, and UFSAR. The reload analyses considered changes relative to Cycle 22 which included cycle specific reload core characteristics (i.e., cycle length, loading pattern, power distributions, etc.) and changes in the unit itself that could impact reload analyses.

The Cycle 23 RAR was prepared to document changes in nuclear, thermal-hydraulic, and mechanical design of the reactor core. As such, the reload report provides the bases for the operation of the Cycle 23 fuel design. All analyses and assessments were performed using NRC approved methodologies. No Technical Specification changes were required to implement the Cycle 23 reload.

EC73060 Cycle 23 Reload Process Applicability Determination (PAD) identified the following adverse changes. Each of the PAD adverse changes are evaluated and are shown to be acceptable.

- PAC exception for the LOCA Unrodded Pin Census – A bounding core pin census was used in the bounding Large Break LOCA (LBLOCA) core-wide oxidation analysis performed for NGF, but the core pin census did not bound the Cycle 23 specific data. Therefore, the limiting LBLOCA case from the Analysis of Record (AOR) was evaluated using the Cycle 23 specific core pin census data. The comparison of results demonstrated that the bounding LBLOCA AOR relevant results remained bounding compared to the Cycle 23 specific core pin census data.
- PAC exception for the LOCA Hot Rod Minimum Pin-to-Box Ratio – The limiting PAC value for the minimum pin-to-box factor for the fuel assembly containing the hot rod for LBLOCA must be less than or equal to the cycle-specific value; however, the Cycle 23 value is below the PAC limit. Therefore, Cycle 23 specific analyses of the limiting case from the LBLOCA AOR were performed using the Cycle 23 minimum hot rod pin-to-box factor, which is more limiting than the value used in the bounding LBLOCA analysis for NGF. To compensate for the loss of margin, the Cycle 23 specific value for the maximum integrated radial peaking factor was also implemented, which is less limiting than the value used in the bounding LBLOCA analysis for NGF. The net result was a reduction in the peak cladding temperature (PCT) for the limiting LBLOCA case.
- PAC exception for the Radial Power Falloff – The Radial Fall-Off (RFO) curve for both non-Integral Fuel Burnable Absorber (IFBA) and IFBA fuel was found to violate the limiting PAC values. The RFO violations for non-IFBA fuel were successfully dispositioned with cycle-specific FATES3B calculations; however, the RFO violations for IFBA fuel could not be dispositioned with cycle-specific FATES3B calculations. As a result, Cycle 23-specific FATES3B files with revised fuel performance initial conditions were generated for LBLOCA analysis. These FATES3B files were shown to be bounded by the AOR fuel performance analysis for the implementation of NGF up to a burnup at the knee in the RFO curve. Since LBLOCA analyses are performed up to the knee in the RFO curve the RFO curve PAC violation has no impact on the LBLOCA AOR. The RFO curve violation does not affect the SBLOCA AOR since the RFO curve violation affects fuel performance initial conditions beyond the knee in the RFO curve and, therefore, beyond the applicability of the SBLOCA AOR.
- Revised CEA drop time – the Cycle 23 Groundrules revised the CEA holding coil delay (HCD) time from 0.6 to 0.8 second, in addition to updating the 5% insertion time from 0.95 to 1.00 second. This results in later reactivity insertion following a reactor trip. There is no impact on the average CEA rod drop time of 3.2 seconds for 90% insertion that is reflected in Technical Specification (TS) 3.1.3.4. Westinghouse document CWTR3-18-47 evaluated the impact of this change on the UFSAR Chapter safety analyses and, for each accident, determined whether or not new analyses were required. All accidents were determined to

be not impacted/bounded by this change or evaluated/assessed to ensure that all reported fuel failures and post-trip, long term transient system responses remained bounded.

The 10CFR50.59 evaluation demonstrated that this change is acceptable. Each of the 8 questions specifically addressed the proposed change and associated impacts.

**Is the validity of this Evaluation dependent on any other change?** ☐ **Yes** ☒ **No**

**If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.**

______________________________________________________________________

______________________________________________________________________

**Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval?** ☐ **Yes** ☒ **No**

**Preparer[2]:** Ben Harvey / See EC73060 / Entergy Services, LLC / Fuels / 12-13-2018

Marcel Provensal / See EC73060 / EOI / WF3 Design Engineering / 12-13-2018
Name (print) / Signature / Company / Department / Date

**Reviewer[2]:** William Steelman / See EC 73060 / EOI / WF3 Design Engineering / 01-16-2019
Name (print) / Signature / Company / Department / Date

**Independent Review[3]:** Peter LeBlond / See EC 73060 (P2E 7.017)/ LeBlond & Associates, LLC / Contractor / 01-09-2019
Name (print) / Signature / Company / Department / Date

**Responsible Manager Concurrence:** Scott C. Stanchfield / See EC 73060 / Entergy Services, LLC / Fuels / 01-18-2019
Name (print) / Signature / Company / Department / Date

**50.59 Program Coordinator Concurrence:** Remy Devoe / See EC 73060 / EOI / WF3 Licensing / 01-14-2019
Name (print) / Signature / Company / Department / Date

**OSRC:** Ran Gilmore / See EC 73060 / 03-08-2019
Chairman's Name (print) / Signature / Date [GGNS P-33633, P-34230, & P-34420; W3 P-151]

19-08
OSRC Meeting #

[2] Either the Preparer or Reviewer will be a current Entergy employee.
[3] If required by Section 5.1[3].

## II. <u>50.59 EVALUATION</u> [10 CFR 50.59(c)(2)]

**Does the proposed Change being evaluated represent a change to a method of evaluation <u>ONLY</u>? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.** ☐ Yes ☒ No

**Does the proposed Change**:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

   BASIS:

   This criterion is concerned with identifying the accidents that have been evaluated in the UFSAR that are affected by the proposed activity. Then a determination is made as to whether the frequency of these accidents occurring would be more than minimally increased.

   The PAC parameter exceptions for the LOCA Unrodded Pin Census, LOCA Hot Rod Minimum Pin-to-Box Ratio, and Radial Power Falloff are used to determine the accident results and severity and do not initiate any accident or transient.

   The revised CEA drop time has no impact on accident frequency since it only impacts the time of reactivity insertion following a reactor trip. The revised CEA drop time also does not change the frequency of a stuck rod. This change does not impact any event initiators and does not initiate any accident or transient.

   The reload analysis evaluates the impact of reload related parameters on the severity of the accident to ensure the results remain within predetermined limits. The adverse changes identified in the EC73060 PAD have no impact on the frequency of occurrence.

   Therefore, the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

The term “malfunction of an SSC important to safety” refers to the failure of structures, systems and components (SSCs) to perform their intended design functions-including both nonsafety-related and safety-related SSCs.

The Cycle 23 fuel and core designs do not degrade the ability of any safety-related system or nonsafety-related system to perform its intended design functions, nor will these changes decrease the reliability of said systems. Instrumentation accuracy or response characteristics are not impacted by this change. None of the failure modes assumed in the UFSAR are impacted by this change.

The revised CEA drop time does not initiate any malfunction of an SSC. The revised CEA drop time also does not increase the frequency of a stuck rod. This change has no impact on malfunction frequency since it only impacts the time of reactivity insertion following a reactor trip.

All equipment important to safety will function in the same manner with the Cycle 23 reload core as with the previous core. There is no characteristic of the Cycle 23 core, with the Batch HH reload assemblies, that would increase the probability of a malfunction of equipment important to safety. The revised CEA drop time and Cycle 23 reload core do not impact any of the failure modes assumed in the UFSAR, nor do they introduce new failure modes.

There are several 10 CFR 50.46 issues associated with the Cycle 23 reload. However, since there is not a 50.46 evaluation form the following three issues will be addressed in the 50.59 evaluation herein.

A bounding core pin census was used in the bounding Large Break LOCA (LBLOCA) core-wide oxidation analysis performed for NGF. The core pin census did not bound the Cycle 23 specific data; therefore, the limiting LBLOCA case from the Analysis of Record (AOR) was evaluated using the Cycle 23 specific core pin census data. The comparison of results demonstrated that the bounding LBLOCA AOR core-wide oxidation (CWO) result remained bounding compared to the Cycle 23 specific core pin census data. The change in core pin census does not impact the maximum local cladding oxidation (MCO). The AOR MCO has been confirmed to remain bounding. Since the AOR remained bounding, there is no impact on any of the failure modes assumed in the UFSAR, nor is a new failure mode introduced.

The limiting PAC value for the minimum pin-to-box factor for the fuel assembly containing the hot rod for LBLOCA must be less than or equal to the cycle-specific value; however, the Cycle 23 value is below the PAC limit. Therefore, Cycle 23 specific analyses of the limiting case from the LBLOCA AOR were performed using the Cycle 23 minimum hot rod pin-to-box factor, which is more limiting than the value used in the bounding LBLOCA analysis for NGF. To compensate for the loss of margin, the Cycle 23 specific value for the maximum integrated radial peaking factor was also implemented, which is less limiting than the value used in the bounding LBLOCA analysis for NGF. The net result was a reduction in the peak cladding temperature (PCT) for the limiting LBLOCA case. Since the cycle specific analysis is bounded by the AOR, there is no impact on any of the failure modes assumed in the UFSAR, nor is a new failure mode introduced.

The Radial Fall-Off (RFO) curve for both non-IFBA and IFBA fuel was found to violate the limiting PAC values during the confirmation of the applicability of the bounding fuel performance analysis for the implementation of NGF to Cycle 23 with the thermal conductivity degradation (TCD) allowance. Cycle 23-specific FATES3B files with revised fuel performance initial conditions were generated for LBLOCA analysis. These FATES3B files were shown to be bounded by the AOR fuel performance analysis for the implementation of NGF up to a burnup at the knee in the RFO curve. Since LBLOCA analyses are performed up to the knee in the RFO curve the RFO curve PAC violation has no impact on the LBLOCA AOR. Of the three PAC exceptions for Cycle 23 described only the RFO curve violation for IFBA fuel is applicable to Small Break LOCA (SBLOCA) analyses. The minimum hot rod pin-to-box factor and the core pin census are not used in SBLOCA analyses. The RFO curve violation does not affect the SBLOCA AOR since the RFO curve violation affects fuel performance initial conditions beyond the knee in the RFO curve and, therefore, beyond the applicability of the SBLOCA AOR. Therefore, since the AOR for LBLOCA and SBLOCA are not impacted, there is no impact on any of the failure modes assumed in the UFSAR, nor is a new failure mode introduced.

Therefore, the likelihood of occurrence of a malfunction of an SSC important to safety is not increased due to the Cycle 23 core reload.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators. All of the adverse conditions were evaluated to ensure their dose consequences did not increase above their limits as discussed below.

Results from cycle-specific analyses for the three PAC exceptions showed that they were either bounded by the limiting LOCA AOR or had no impact and were confirmed to remain bounded by the LOCA AOR. Since the AOR remained bounding, there is no impact to the dose consequences for these events.

Westinghouse document CWTR3-18-47 evaluated the impact of the revised CEA drop time on the UFSAR Chapter 15 safety analyses and, for each accident, determined whether or not new analyses were required for this change. For the accidents which were reanalyzed and for those which were determined to be bounded, the UFSAR Chapter 15 safety analyses remain acceptable and meet the requirements with the inclusion of the revised CEA drop time. Thus, there are no impacts on reported fuel failures and post-trip, long term transient system responses remain bounded. Since none of the inputs to the radiological dose calculations are impacted by this change, there is no impact on radiological doses/consequences.

Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators.

The Waterford-3 Cycle 23 reload safety analyses were performed to assure that acceptance criteria are met for fuel performance, thermal-hydraulic performance, LOCA Emergency Core Cooling System (ECCS) performance and non-LOCA transient response. The revised CEA drop time has no impact on the core design, thermal-hydraulics, and fuel rod design analyses. These analyses confirm that the Cycle 23 core can be operated safely and can be expected to meet license requirements for accident response. The function and duty of SSCs important to safety as assumed in the safety analyses is not altered. The Cycle 23 analyses do not place greater reliance on any specific plant SSC to perform a safety function. No changes in the assumptions concerning equipment availability or failure modes have been made to implement Cycle 23.

Therefore, the proposed change does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

This criterion is concerned with creating the possibility for accidents of similar frequency and significance to those already included in the licensing basis for the facility.

Neither the Cycle 23 reload core nor the revised CEA drop time create any new interactions, directly or indirectly, which could cause an accident of a different type. These changes do not result in changes to the radiological release rate/duration, do not create new release mechanisms, and do not impact radiation release barriers. There are no new system interactions or connections associated with the Cycle 23 core reload.

Therefore, the proposed change does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

This criterion is concerned with malfunctions that involve initiators or failures whose effects are not bounded by those explicitly described in the UFSAR.

The Cycle 23 reload core and revised CEA drop time do not impact the failure modes assumed in the UFSAR. Equipment important to safety will function in the same manner with the Cycle 23 core as with the Cycle 22 core. The impact of changes in core characteristics on any parameter that would affect the function of equipment important to safety has been accounted for in the analyses applicable for Cycle 23.

The Waterford-3 testing and verification program ensures that all required calibrations and setpoint changes resulting from the Cycle 23 reload design are performed. There are no new modes of failure associated with any of the changes for Cycle 23. No changes in the failure modes of the equipment important to safety were assumed in the Cycle 23 core design or fuel mechanical analyses. No changes due to the Cycle 23 reload analysis will significantly alter the way in which Waterford-3 operates.

Based on the above, the proposed change does not create a possibility of a malfunction of equipment important to safety having a different result than any previously evaluated in the UFSAR.

7. Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

This criterion is concerned with fission product barriers and the critical design information that supports their continued integrity.

The revised CEA drop time impacts the time at which reactivity is inserted into the core; however, the TS 3.1.3.4 limit of 3.2 seconds for 90% insertion is not impacted. The time of reactivity insertion into the core does impact the fuel and therefore the cladding (one of the three fission product barriers); however, the CEA insertion times are not design basis limits but rather inputs. Westinghouse document CWTR3-18-47 evaluated the impact of the revised CEA drop time on the UFSAR Chapter 15 safety analyses and, for each accident, determined whether or not new analyses were required for this change. For the accidents which were reanalyzed and for those which were determined to be bounded, the UFSAR Chapter 15 safety analyses remain acceptable and meet the requirements with the inclusion of the revised CEA drop time. Thus, there are no impacts on reported fuel failures and post-trip, long term transient system responses remain bounded. No design basis limits are exceeded or altered by this change. The revised CEA drop time has no impact on the core design, thermal-hydraulics, and fuel rod design analyses.

The Waterford-3 Cycle 23 reload safety analyses were performed to assure that acceptance criteria are met for fuel performance, thermal-hydraulic performance, LOCA ECCS performance and non-LOCA response. There are three exceptions applicable to the LOCA analysis from the PAC for Cycle 23 which can directly affect PCT, MCO, and CWO. Results from cycle-specific analyses for these three PAC exceptions showed that they were either bounded by the limiting LOCA AOR or had no impact and were confirmed to remain bounded by the LOCA AOR. All of these analyses confirm that the core can be operated safely and can be expected to meet license requirements for accident response. The Cycle 23 reload safety analyses were performed to demonstrate compliance with the existing design basis limits for the fuel cladding, RCS pressure boundary, and containment. All events have been evaluated in the reload analysis to assure that they meet their respective criterion for Cycle 23.

Based on the above, the proposed change does not result in a design basis limit for a fission product barrier being exceeded or altered.

8. Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

Method of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC [NEI 96-07 Revision 1, Section 3.10]. In accordance with Technical Specification 6.9.1.11.1, the Cycle 23 core was designed and evaluated using NRC approved analysis methodology under an approved quality assurance program. No new methodologies were required to verify that previous safety analyses are applicable to Cycle 23 or to perform reanalysis of any events.

All analyses performed in support of the revised CEA drop time are consistent with the methodologies described in the UFSAR. As such, no methods of evaluation are affected by this change.

Therefore, the proposed change does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

**If any of the above questions is checked “Yes,” obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.**

## I. OVERVIEW / SIGNATURES[1]

**Facility:** Waterford 3 **Evaluation # / Rev. #:** 2018-03 / 0

**Proposed Change / Document:** EC-78061, Turbine Valve Test Extension

**Description of Change:** EC-78061 evaluates a one-time extension of the Technical Requirements Manual (TRM) 4.3.4.2.a turbine valve testing in order to improve summer reliability. The current late date will be extended from 7/18/2018 to 12/18/2018.

TRM 4.3.4.2.a states:
At least once every 184 days, (under direct observation), each of the following valves is cycled through at least one complete cycle from the running position:

1. Four high pressure throttle valves.
2. Four high pressure governor valves.
3. Six low pressure reheat stop valves.
4. Six low pressure reheat intercept valves.

This change validated that the turbine missile ejection probability remains within the regulatory requirements and is consistent with the analysis described in UFSAR Sections 3.5.1.3 and 10.2.3. This change will extend the TRM 4.3.4.2.a turbine valve testing late date from 7/18/2018 to 12/18/2018.

LBDCR 18-010 implements the one-time extension to TRM 4.3.4.2.a turbine valve testing moving the late date from 7/18/2018 to 12/18/2018.

LBDCR 18-011 removes the one-time extension to TRM 4.3.4.2.a turbine valve testing after 12/18/2018 to restore the TRM information.

**Summary of Evaluation:**
This change addresses extending the Technical Requirements Manual (TRM) 4.3.4.2.a turbine valve testing late date from 7/18/2018 to 12/18/2018. This change was performed by using the UFSAR Section 3.5.1.3.7 probability calculation methodology and updating for a TRM 4.3.4.2.a turbine valve testing interval of 11 months and plant specific turbine inspection data. The use of the plant specific turbine inspection data reduced the probability to below the UFSAR values.

There are two NRC requirements that Waterford 3 must meet for this change to be acceptable. The first comes from Regulatory Guide 1.115 [References 5]. Regulatory Guide 1.115 page 4 states that the probability of failure of an essential Structure, System, or Component (SSC) because of turbine missiles is calculated from equation $P_{total} = P_1 \times P_2 \times P_3$. Where $P_1$ is the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing. $P_2$ is the probability of ejected missiles perforating intervening barriers and striking essential SSCs. $P_3$ is the probability of essential SSCs that are struck failing to perform their safety functions. $P_{total}$ is limited to less than $1x10^{-7}$ per year, which the NRC staff considers to be an acceptable risk rate for the loss of an essential SSC from a single event. Thus, the first requirement is $P_{total}$ is less than $1x10^{-7}$ per year.

[1] The printed name, company, department, and date must be included on the form. Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

**Summary of Results Table**

| Description | $P_{total}$ – UFSAR Section 3.5.1.3.7 | $P_{total}$ – EC-78061 Using plant specific turbine inspection data | $P_{total}$ - Acceptance Limit |
|---|---|---|---|
| Current TRM 4.3.4.2.a turbine valve testing interval (184 days) | $7.48 \times 10^{-8}$ per year | $3.08 \times 10^{-8}$ per year | $< 1 \times 10^{-7}$ per year |
| EC78061 TRM 4.3.4.2.a turbine valve testing extension to 11 months | NA | $3.36 \times 10^{-8}$ per year | $< 1 \times 10^{-7}$ per year |

UFSAR Section 3.5.1.3.7 provides the $P_{total}$ results for the current TRM 3/4.3.4 surveillance frequencies. The UFSAR Section 3.5.1.3.7 total probability of strike damage, $P_{total}$, is $7.48 \times 10^{-8}$ per year. This change has determined that that the new $P_{total}$ is $3.36 \times 10^{-8}$ per year [Reference 11] which remains below the NRC requirement of $1 \times 10^{-7}$ per year. This $P_{total}$ is also below the UFSAR Section 3.5.1.3.7 value because the analysis performed used plant specific turbine inspection times to calculate the probabilities.

The second NRC requirement that must be met is the change in probability of a component level malfunction must be less than a factor of 2 increase. This means that the new $P_{total}$ (or $P_1$) must be validated to be less than a factor of 2 increase. Since the component level malfunction is a potential initiator of some accidents, the new $P_{total}$ (or $P_1$) must be validated to be less than a factor of 10% increase. The factor of 2 increase is from 50.59 Question #2 (Does the Activity Result in More Than a Minimal Increase in the Likelihood of Occurrence of a Malfunction of an SSC Important to Safety?) which clarified in NEI 96-07 Section 4.3.2 [Reference 12]. NEI 96-07 Section 4.3.2 Example 8 states that if the change in likelihood of occurrence of a malfunction is calculated in support of the evaluation and increases by more than a factor of two require NRC approval. NEI 96-07 Section 4.3.1 Example 3 states that an increase in the pre-change accident or transient frequency exceeding 10% would require prior NRC approval.

UFSAR Section 3.5.1.3.7 states that the total probability of strike damage, $P_{total}$, is $7.48 \times 10^{-8}$ per year. Using updated data to calculate $P_1$, the new $P_{total}$ is $3.08 \times 10^{-8}$ per year based on a six month valve test frequency. The extension in turbine valve testing frequency from six months to eleven months increases $P_{total}$ to $3.36 \times 10^{-8}$ per year. Though this change increases the strike damage probability by $2.8 \times 10^{-9}$, $P_{total}$ is less than the acceptance limit ($3.36 \times 10^{-8} < 1 \times 10^{-7}$). In addition, the probability increases less than 10% (9.1%) from $3.08 \times 10^{-8}$ per year to $3.36 \times 10^{-8}$ per year when extending the turbine valve testing frequency from six months to eleven months. This means that this change is not a factor of two increase in the likelihood of a malfunction on the component level and is not more than a 10% increase in the frequency of occurrence of an accident. This change remains within the minimal increase in likelihood of occurrence requirement. The new $P_{total}$ is below the UFSAR Section 3.5.1.3.7 value because the analysis performed used plant specific turbine inspection times to calculate the probabilities. In addition, the probabilistic analysis performed a one to one comparison using the plant specific turbine inspection times. The $P_{total}$ for turbine valve testing time of 6 months is $3.08 \times 10^{-8}$ per year and the 11-month probability is $3.36 \times 10^{-8}$ per year. Thus, the proposed change to extend the turbine valve test frequency one time from 184 days to 334 days is acceptable to implement under 10CFR50.59.

The 10CFR50.59 evaluation demonstrated that this change is acceptable. Each of the 8 questions specifically addressed the proposed change and associated impacts. The TRM 3/4.3.4 change to extend the late date of 7/18/2018 to 12/18/2018 is acceptable.

**References:**

1. Waterford 3 Updated Final Safety Analysis Report
2. Waterford 3 Technical Requirements Manual
3. NRC Waterford 3 Operating License Amendment 103, Turbine Overspeed Protection Relocation, March 2, 1995
4. Regulatory Guide 1.115 Revision 1, Protection Against Low Trajectory Turbine Missiles, July 1977
5. Regulatory Guide 1.115 Revision 2, Protection Against Turbine Missiles, January 2012
6. Nuclear Electric Insurance Limited (NEIL) – Loss Control Standard 3
7. Waterford 3 Calculation 3T1-18, Turbine Missiles, March 1982
8. WCAP-16501-P, Extension of Turbine Valve Test Frequency Up to 6 Months for BB-296 Siemens Power Generation (Westinghouse) Turbines with Steam Chests, February 2006
9. Westinghouse Technical Memo TM-94246, Turbine Valve Testing Frequency for Entergy Operations Waterford Station, October 1994
10. ER-W3-2006-0164-000, Revise TRM 3/4.3.4 to Extend LP Turbine Disc Inspection Interval, August 31, 2006
11. CWTR3-18-27, Transmittal of the Low Pressure Turbine Valve Test Interval Evaluation for Waterford 3, July 11, 2018
12. NEI 96-07 Revision 1, Guidelines for 10CFR50.59 Implementation, November 2000
13. NUREG-0787 Supplement 4, NRC Safety Evaluation Report for Waterford 3, October 1982
14. EC-65610, TRM 3.3.4 Turbine Valve Testing One-Time Extension
15. CWTR3-16-14, Transmittal of the Low Pressure Turbine Valve Test Interval Evaluation for Waterford 3, July 12, 2016

**Is the validity of this Evaluation dependent on any other change?** ☐ **Yes** ☒ **No**

**If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.**

**Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval?** ☐ **Yes** ☒ **No**

| | |
|---|---|
| **Preparer[2]:** | Dale Gallodoro / See EC-78061 / SMI / Design Engineering / 7-10-2018 |
| | Name (print) / Signature / Company / Department / Date |
| **Reviewer[2]:** | Harry LeBlanc / See EC-78061 / SMI / Systems Engineering / 7-10-2018 |
| | Name (print) / Signature / Company / Department / Date |
| **Reviewer[2]:** | Joseph Lanci / See EC-78061 / EOI / Systems Engineering / 7-12-2018 |
| | Name (print) / Signature / Company / Department / Date |
| **Independent Review[3]:** | N/A |
| | Name (print) / Signature / Company / Department / Date |
| **OSRC:** | Ran Gilmore / See EC-78061 / 7-12-2018 |
| | Chairman's Name (print) / Signature / Date [GGNS P-33633, P-34230, & P-34420; W3 P-151] |
| | W3-18-13 |
| | OSRC Meeting # |

2 Either the Preparer or Reviewer will be a current Entergy employee.
3 If required by Section 5.1[3].

## II. 50.59 EVALUATION [10 CFR 50.59(c)(2)]

**Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.**

☐ Yes
☒ No

**Does the proposed Change**:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

This change addresses extending the Technical Requirements Manual (TRM) 4.3.4.2.a turbine valve testing late date from 7/18/2018 to 12/18/2018. The UFSAR was reviewed to identify which accidents previously evaluated could be impacted by the turbine admission valves. The accident analyses that are already included in the UFSAR are as follows:

UFSAR Section 15.1.1.3 (Increased Main Steam Flow) is a moderate frequency incident. The increase in heat removal by the steam generators as a result of increased main steam flow is defined as any rapid increase in steam generator steam flow, other than a steam line rupture, without a turbine trip. The limiting failure for this event is the failure in the Steam Bypass System which could result in an opening of one of the turbine bypass valves.

UFSAR Section 15.1.2.3 (Increased Main Steam Flow with a Concurrent Loss of Offsite Power) is classified as an infrequent incident. For this event, it is conservatively assumed that the increased heat removal due to excess main steam flow uses up all thermal margin initially preserved by CPC's and/or COLSS. Then a Loss of Offsite Power is modeled to further reduce the thermal margin.

UFSAR Section 15.2.1.1 (Loss of External Load) is classified as a moderate frequency incident. A loss of external load results in a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves.

UFSAR Section 15.2.1.2 (Turbine Trip) is classified as a moderate frequency incident. A turbine trip can be produced by a turbine overspeed.

UFSAR Section 15.2.2.1 (Loss of External Load with a Concurrent Single Failure of an Active Component) is classified as an infrequent incident. A loss of external load results in a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves with a concurrent single failure of an active component.

UFSAR Section 15.2.2.2 (Turbine Trip with A Concurrent Single Failure of an Active Component) is classified as an infrequent incident. A turbine trip can be produced by a turbine overspeed.

The determination of the impact on the event frequencies uses guidance provided in NEI 96-07 [Reference 12] Section 4.3.1. This section states:

> During initial plant licensing, accidents were typically assessed in relative frequencies. Minimal increases in frequency resulting from subsequent licensee activities do not significantly change the licensing basis of the facility and do not impact the conclusions reached about acceptability of the facility design.

The TRM 4.3.4.2.a turbine valve testing will impact the probability of failure of an essential SSC because of turbine missile. This change in probability will be used to determine the impact of the initial plant licensing frequencies.

The Waterford 3 original probability of failure of an essential SSC because of turbine missiles is listed in UFSAR Section 3.5.1.3.6 as $9.4 \times 10^{-8}$ per year (design overspeed $2.6 \times 10^{-8}$ plus destructive overspeed $6.8 \times 10^{-8}$). The turbine missile probabilities were recalculated in 1994 and 2006 using newer values of valve failure rates and are listed in UFSAR Section 3.5.1.3.7. The UFSAR Section 3.5.1.3.7 probability of failure of an essential SSC because of turbine missiles is $7.48 \times 10^{-8}$ per year (2006 value). The EC-78061 analysis used the same methodology as the UFSAR Section 3.5.1.3.7 and obtained a turbine missile strike failure rate of $3.08 \times 10^{-8}$ per year with a turbine valve test frequency of 6 months and $3.36 \times 10^{-8}$ per year with a turbine valve test frequency of 11 months. These $P_{total}$'s are still below the UFSAR Section 3.5.1.3.7 value of $7.48 \times 10^{-8}$ per year and the Regulatory Guide 1.115 acceptance criteria of $1 \times 10^{-7}$ per year. NEI 96-07 Section 4.3.1 Example 3 states that an increase in the pre-change accident or transient frequency exceeding 10% would require prior NRC approval. Because the probability increases less than 10% from $3.08 \times 10^{-8}$ per year to $3.36 \times 10^{-8}$ per year when extending the turbine valve testing frequency from six months to eleven months and because the frequency of occurrence remains below the acceptance limit of $1 \times 10^{-7}$ per year acceptance limit, this change remains within the minimal increase in frequency of an accident requirement. Thus, there is not more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

This change addresses extending the Technical Requirements Manual (TRM) 4.3.4.2.a turbine valve testing late date from 7/18/2018 to 12/18/2018. This change was performed by using the UFSAR Section 3.5.1.3.7 probability calculation methodology and updating for a TRM 4.3.4.2.a turbine valve testing time of 11 months and plant specific turbine inspection data. The use of the plant specific turbine inspection data reduced the probability to below the UFSAR values. The change in likelihood of occurrence of a malfunction of the turbine valves due to the turbine valve surveillance extension is proportional to the change in strike damage frequency. There are two (2) NRC requirements that must be met for this change to be acceptable. The first comes from Regulatory Guide 1.115 [Reference 5]. Regulatory Guide 1.115 page 4 states that the probability of failure of an essential Structure, System, or Component (SSC) because of turbine missiles is calculated from equation $P_{total} = P_1 \times P_2 \times P_3$. Where $P_1$ is the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing. $P_2$ is the probability of ejected missiles perforating intervening barriers and striking essential SSCs. $P_3$ is the probability of essential SSCs that are struck failing to perform their safety functions. $P_{total}$ is limited to less than $1 \times 10^{-7}$ per year, which the NRC staff considers to be an acceptable risk rate for the loss of an essential SSC from a single event. Thus, the first requirement is $P_{total}$ is less than $1 \times 10^{-7}$ per year.

Regulatory Guide 1.115 Table 1 (Summary of the NRC Criteria for Turbine Missiles) provides the regulatory acceptance criteria. Regulatory Guide 1.115 Table 1 shows that the NRC assumes the $P_2 \times P_3 = 10^{-2}$ per year for unfavorably oriented turbines. The $P_2 \times P_3$ value is an NRC assumed conservative value; the site-specific calculation for $P_2 \times P_3$ is $2.128 \times 10^{-3}$. The NRC concluded in the original licensing of Waterford 3 that the turbine generator placement and orientation is unfavorable with respect to the plant containment building and other vital areas. This configuration places the Reactor Building, Reactor Auxiliary Building (RAB), and control room within the path of both the high and low trajectory turbine missile. With $P_2 \times P_3 = 10^{-2}$ per year, the NRC acceptance criteria for $P_1$ is $10^{-5}$ per year. This means that if $P_1$ is less than $10^{-5}$ per year, then $P_{total}$ will be less than $1x10^{-7}$ per year.

In simplistic terms, the probability of missile generation, $P_1$, is the sum of the probability of a given turbine speed multiplied by the conditional probability of generation of a missile if the turbine is at that speed. The turbine speed components generally consist of the running speed, design overspeed, and destructive overspeed with the conditional probability of generation of a missile being a function of rotor durability, rotor inspection frequency, turbine valve failure rates, turbine valve test frequency and generator trip frequency.

**Summary of Results Table**

| Description | $P_{total}$ – UFSAR Section 3.5.1.3.7 | $P_{total}$ – EC-78061 Using plant specific turbine inspection data | $P_{total}$ - Acceptance Limit |
|---|---|---|---|
| Current TRM 4.3.4.2.a turbine valve testing interval (184 days) | $7.48 \times 10^{-8}$ per year | $3.08 \times 10^{-8}$ per year | $< 1 \times 10^{-7}$ per year |
| EC78061 TRM 4.3.4.2.a turbine valve testing extension to 11 months | NA | $3.36 \times 10^{-8}$ per year | $< 1 \times 10^{-7}$ per year |

UFSAR Section 3.5.1.3.7 provides the $P_{total}$ results for the current TRM 3/4.3.4 surveillance frequencies. The UFSAR Section 3.5.1.3.7 total probability of strike damage, $P_{total}$, is $7.48 \times 10^{-8}$ per year. This change has determined that that the new $P_{total}$ is $3.36 \times 10^{-8}$ per year [Reference 11] which remains below the NRC requirement of $1x10^{-7}$ per year. This $P_{total}$ is also below the UFSAR Section 3.5.1.3.7 value because the analysis performed used plant specific turbine inspection times to calculate the probabilities.

The second NRC requirement that must be met is the change in likelihood of a component level malfunction must be less than a factor of 2 increase. This means that the new $P_{total}$ (or $P_1$) must be validated to be less than a factor of 2 increase. The factor of 2 increase is from NEI 96-07 Section 4.3.2 [Reference 12]. NEI 96-07 Section 4.3.2 Example 8 states that if the change in likelihood of occurrence of a malfunction is calculated in support of the evaluation and increases by more than a factor of two then NRC approval is required.

The UFSAR Section 3.5.1.3.7 total probability of strike damage, $P_{total}$, is $7.48 \times 10^{-8}$ per year. The new $P_{total}$ is $3.08 \times 10^{-8}$ per year for a six month turbine valve test interval and $3.36 \times 10^{-8}$ per year for an eleven month turbine valve test interval. Though this change increases strike damage probability by $2.8 \times 10^{-9}$, $P_{total}$ is still less than the acceptance limit ($3.36 \times 10^{-8} < 1 \times 10^{-7}$). In addition, the probability increases less than 10% from $3.08 \times 10^{-8}$ per year to $3.36 \times 10^{-8}$ per year when extending the turbine valve testing frequency from six months to eleven months. This means that this change is not a factor of two increase. This change remains within the minimal increase in likelihood of occurrence requirement. The new $P_{total}$ is below the UFSAR Section 3.5.1.3.7 value because the analysis performed used plant specific turbine inspection times to calculate the probabilities. In addition, the probabilistic analysis performed a one to one comparison using the plant specific turbine inspection times. Thus, the plant specific probability does not increase by more than a factor of 10% when increasing the turbine valve testing time to 11 months. This change remains within the minimal increase in frequency of an accident or likelihood of a malfunction requirement.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:
When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators.

This change addresses extending the Technical Requirements Manual (TRM) 4.3.4.2.a turbine valve testing late date from 7/18/2018 to 12/18/2018. The UFSAR accidents previously evaluated which could be impacted by the turbine admission valves are as follows:

UFSAR Section 15.1.1.3 (Increased Main Steam Flow) event causes an increase in heat removal by the steam generators as a result of increased main steam flow. The increased main steam flow is defined as any rapid increase in steam generator steam flow without a turbine trip. The limiting failure for this event is the failure in the steam bypass system which could result in an opening of one of the turbine bypass valves. With the steam bypass being the limiting failure, the extension of the TRM 4.3.4.2.a turbine valve testing will have no impact on these event consequences. Thus, no adverse impact.

UFSAR Section 15.1.2.3 (Increased Main Steam Flow with a Concurrent Loss of Offsite Power) event is conservatively assumed that the increased heat removal due to excess main steam flow uses up all thermal margin initially preserved by CPC's and/or COLSS. Then a Loss of Offsite Power is modeled to further reduce the thermal margin. The excess steam demand portion of this transient could be caused any number of failures (steam bypass valve, atmospheric dump valve, steam leak, turbine controls). This event is conservatively initiates an increased steam flow (cause is not important) to reduce thermal margin just above the CPC trip setpoint. For this event, turbine valves cannot cause an excess steam demand more severe than that analyzed in this section. Thus, no adverse impact.

UFSAR Section 15.2.1.1 (Loss of External Load) event results in a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves. This event is a heat up transient which causes a primary and secondary pressure transient, so a rapid closure of the turbine stop valves produces the most adverse consequences. The TRM 4.3.4.2.a turbine valve testing extension cannot cause a more rapid closure of the turbine stop valves. The turbine valve testing is intended to validate free and smooth motion to ensure no buildup of deposits on the shafts. This means for heat up events, the potential slower motion would be advantageous and make the potential radiological consequences less adverse.

In addition, the loss of external load radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in UFSAR Section 15.1.2.4.5. The loss of external load radiological consequences will continue to be bounded by the UFSAR Section 15.1.2.4.5 event.

UFSAR Section 15.2.1.2 (Turbine Trip) is classified as a moderate frequency incident. A turbine trip can be produced by a turbine overspeed. This event is a heat up transient which causes a primary and secondary pressure transient, so a rapid closure of the turbine stop valves produces the most adverse consequences. The TRM 4.3.4.2.a turbine valve testing extension cannot cause a more rapid closure of the turbine stop valves. The turbine valve testing is intended to validate free and smooth motion to ensure no buildup of deposits on the shafts. This means for heat up events, the potential slower motion would be advantageous and make the potential radiological consequences less adverse.

In addition, the turbine trip radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in UFSAR Section 15.1.2.4.5. The turbine trip radiological consequences will continue to be bounded by the UFSAR Section 15.1.2.4.5 event.

UFSAR 15.2.2.1 (Loss of External Load with a Concurrent Single Failure of an Active Component) results in a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves with a concurrent single failure of an active component. This event is a heat up transient which causes a primary and secondary pressure transient, so a rapid closure of the turbine stop valves produces the most adverse consequences. The TRM 4.3.4.2.a turbine valve testing extension cannot cause a more rapid closure of the turbine stop valves. The turbine valve testing is intended to validate free and smooth motion to ensure no buildup of deposits on the shafts. This means for a heat up events, the potential slower motion would be advantageous and make the potential radiological consequences less adverse.

In addition, the loss of external load radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in UFSAR Section 15.1.2.4.5. The loss of external load radiological consequences will continue to be bounded by the UFSAR Section 15.1.2.4.5 event.

UFSAR Section 15.2.2.2 (Turbine Trip with A Concurrent Single Failure of an Active Component) is classified as an infrequent incident. A turbine trip can be produced by a turbine overspeed. This event is a heat up transient which causes a primary and secondary pressure transient, so a rapid closure of the turbine stop valves produces the most adverse consequences. The TRM 4.3.4.2.a turbine valve testing extension cannot cause a more rapid closure of the turbine stop valves. The turbine valve testing is intended to validate free and smooth motion to ensure no buildup of deposits on the shafts. This means for heat up events, the potential slower motion would be advantageous and make the potential radiological consequences less adverse.

In addition, the turbine trip radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in UFSAR Section 15.1.2.4.5. The turbine trip radiological consequences will continue to be bounded by the UFSAR Section 15.1.2.4.5 event.

The accidents previously evaluated in the UFSAR were assessed for potential impacts and no adverse consequences were identified because there is no change in the target set for potential turbine missiles. The proposed change does not adversely impact the capability of the turbine valves to perform their specified functions. Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
When determining which malfunctions represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. This change addresses extending the Technical Requirements Manual (TRM) 4.3.4.2.a turbine valve testing late date from 7/18/2018 to 12/18/2018.

In general, design basis accidents have an event initiator and may assume one active single failure. The design basis accidents have identified their limiting single failures with respect to specific acceptance criteria. This change does not change any of the physical structures, systems, or components. The proposed change also does not create any new system interactions that could cause a malfunction. There are no changes in the potential targets for the potential turbine missiles. This change does not place any greater reliance on any SSC because of the proposed change. The TRM 4.3.4.2.a turbine valve testing extension only changes the time for the testing. The system specified functions and interactions remain unchanged. This means that the existing UFSAR limiting failures remain unchanged.

Therefore, the turbine valve potential failures remain no more adverse than that already analyzed in the UFSAR. The proposed change does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
UFSAR Section 3.5.1.3 (Turbine Missiles) describes failures that could occur in the large steam turbines that could produce large high-energy missiles. The potential for damage to safety related structures, systems and components due to such turbine failure has been evaluated to determine whether additional protection, beyond that inherently provided by existing structural shielding, is required to further reduce the damage probability. UFSAR Section 10.2.3 (Turbine Disk Integrity) describes the turbine materials and potential failure modes. UFSAR Section 15.2.1.2 and 15.2.2.2 already considered a turbine trip due to an overspeed condition.

The generation of turbine missiles and turbine overspeed are already considered within the UFSAR 3.5.1.3, 10.2.3, 15.2.1.2, and 15.2.2.2. This change addresses extending the Technical Requirements Manual (TRM) 4.3.4.2.a turbine valve testing late date from 7/18/2018 to 12/18/2018. A change in interval for turbine valve testing does not create any additional failure mechanisms.
Therefore, an accident of a different type is not possible.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
Regulatory Guide 1.115 [Reference 5] page 4 states that the probability of failure of an essential Structure, System, or Component (SSC) because of turbine missiles is calculated from equation $P_{total} = P_1 \times P_2 \times P_3$. Where $P_1$ is the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing. $P_2$ is the probability of ejected missiles perforating intervening barriers and striking essential SSCs. $P_3$ is the probability of essential SSCs that are struck failing to perform their safety functions.

The $P_2$ probability involves the potential for striking new or different targets. The $P_3$ probability involves the potential for those new targets failing. The $P_2$ and $P_3$ probabilities remain unchanged from those already contained in UFSAR Section 3.5.1.3.7 which means no new targets or failures are predicted. That also means that this change will not result in a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR.

7. Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

This change addresses extending the Technical Requirements Manual (TRM) 4.3.4.2.a turbine valve testing late date from 7/18/2018 to 12/18/2018. The UFSAR accidents previously evaluated which could be impacted by the turbine admission valves are as follows:

UFSAR Section 15.1.1.3 (Increased Main Steam Flow) event causes an increase in heat removal by the steam generators as a result of increased main steam flow. The increased main steam flow is defined as any rapid increase in steam generator steam flow without a turbine trip. The limiting failure for this event is the failure in the steam bypass system which could result in an opening of one of the turbine bypass valves. With the steam bypass being the limiting failure, the extension of the TRM 4.3.4.2.a turbine valve testing will have no impact on these event consequences. Thus, no adverse impact.

UFSAR Section 15.1.2.3 (Increased Main Steam Flow with a Concurrent Loss of Offsite Power) event is conservatively assumed that the increased heat removal due to excess main steam flow uses up all thermal margin initially preserved by CPC's and/or COLSS. Then a Loss of Offsite Power is modeled to further reduce the thermal margin. The excess steam demand portion of this transient could be caused any number of failures (steam bypass valve, atmospheric dump valve, steam leak, turbine controls). This event is conservatively initiates an increased steam flow (cause is not important) to reduce thermal margin just above the CPC trip setpoint. For this event, turbine valves cannot cause an excess steam demand more severe than that analyzed in this section. Thus, no adverse impact.

UFSAR Section 15.2.1.1 (Loss of External Load) event results in a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves. This event is a heat up transient which causes a primary and secondary pressure transient, so a rapid closure of the turbine stop valves produces the most adverse consequences. The TRM 4.3.4.2.a turbine valve testing extension cannot cause a more rapid closure of the turbine stop valves. In addition, UFSAR Section 15.2.1.1.4 (Barrier Performance) states that the loss of external load consequences would be less adverse than those following an UFSAR Section 15.2.1.3 loss of condenser vacuum. This means that no matter the creditable failure of the turbine valves, the barrier consequences would remain bounded by another event. Thus, no adverse impact.

UFSAR Section 15.2.1.2 (Turbine Trip) is classified as a moderate frequency incident. A turbine trip can be produced by a turbine overspeed. This event is a heat up transient which causes a primary and secondary pressure transient, so a rapid closure of the turbine stop valves produces the most adverse consequences. The TRM 4.3.4.2.a turbine valve testing extension cannot cause a more rapid closure of the turbine stop valves. In addition, UFSAR Section 15.2.1.2.4 (Barrier Performance) states that the loss of external load consequences would be less adverse than those following an UFSAR Section 15.2.1.3 loss of condenser vacuum. This means that no matter the creditable failure of the turbine valves, the barrier consequences would remain bounded by another event. Thus, no adverse impact.

UFSAR 15.2.2.1 (Loss of External Load with a Concurrent Single Failure of an Active Component) results in a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves with a concurrent single failure of an active component. This event is a heat up transient which causes a primary and secondary pressure transient, so a rapid closure of the turbine stop valves produces the most adverse consequences. The TRM 4.3.4.2.a turbine valve testing extension cannot cause a more rapid closure of the turbine stop valves. In addition, UFSAR Section 15.2.2.1.4 (Barrier Performance) states that the loss of external load consequences would be less adverse than those following an UFSAR Section 15.2.1.3 loss of condenser vacuum. This means that no matter the creditable failure of the turbine valves, the barrier consequences would remain bounded by another event. Thus, no adverse impact.

UFSAR Section 15.2.2.2 (Turbine Trip with A Concurrent Single Failure of an Active Component) is classified as an infrequent incident. A turbine trip can be produced by a turbine overspeed. This event is a heat up transient which causes a primary and secondary pressure transient, so a rapid closure of the turbine stop valves produces the most adverse consequences. The TRM 4.3.4.2.a turbine valve testing extension cannot cause a more rapid closure of the turbine stop valves. In addition, UFSAR Section 15.2.2.2.4 (Barrier Performance) states that the loss of external load consequences would be less adverse than those following an UFSAR Section 15.2.1.3 loss of condenser vacuum. This means that no matter the creditable failure of the turbine valves, the barrier consequences would remain bounded by another event. Thus, no adverse impact.

The accidents previously evaluated in the UFSAR were assessed for potential impacts and no adverse consequences were identified. Thus, a design basis limit for a fission product barrier as described in the UFSAR will not be exceeded or altered.

8. Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:
The calculational frameworks used for the proposed change are screened out in the PAD where it was concluded that the proposed changes do not adversely affect a method of evaluation that demonstrates intended design functions of an SSC will be accomplished as described in the UFSAR. Therefore, an evaluation for 10CFR50.59(c)(2)(viii) is not required for the proposed change.

**If any of the above questions is checked “Yes,” obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.**

## I. OVERVIEW / SIGNATURES[1]

**Facility:** Waterford 3 Steam Electric Station **Evaluation # / Rev. #:** 2019-01 / 0

**Proposed Change / Document:** EC81569 Technical Requirements Manual 3/4.9.6 Change

**Description of Change:**
Engineering Change (EC) EC81569 [Reference 10] will allow a "one time" exception to Technical Requirements Manual 3/4.9.6 (Refuel Machine) so that new fuel assembly LAHE20 can be safely moved using rigging attached to the refuel machine control element assembly (CEA) beam.

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**Summary of Evaluation:**

EC81569 will allow the refuel machine CEA beam to be used to safely lift a new fuel assembly that is not seated on the bottom core support plate alignment pins. The use of the CEA beam, load cell, and manual rigging will take the place of the refuel machine interlocks. These interlocks are not credited for mitigation in the UFSAR Section 15.7.3.4 fuel handling accident and the operator action is not required to support a design function credited in the safety analysis

UFSAR Section 9.1.4 describes that the fuel handling equipment includes interlocks, travel limiting features, and other protective devices to minimize the possibility of inadvertent damage to a fuel assembly and potential fission product release, resulting from either mishandling or equipment malfunction. UFSAR Section 9.1.4.2.1.1 lists the interlocks associated with the refuel machine. Technical Requirements Manual 3/4.9.6 requires a refuel machine overload cut off limit.

The evaluation demonstrates that this change will not increase the frequency of an accident or likelihood of a malfunction because the equipment used is rated for the application and controls are in place to preclude overload limits from being exceeded. The potential accident and malfunction consequences continue to be bounded by the UFSAR Section 15.7.3.4 fuel handling accident. No new accidents or equipment malfunctions were identified. No new methodologies were utilized.

The 10CFR50.59 evaluation demonstrated that this change is acceptable. Each of the 8 questions specifically addressed the proposed change and associated impacts.

References:
1. Waterford Nuclear Generating Station Unit 3, Updated Final Safety Analysis Report (UFSAR).
2. Waterford Nuclear Generating Station Unit 3, Technical Specifications.
3. Waterford Nuclear Generating Station Unit 3, Technical Requirements Manual.
4. NEI 96-07 Revision 1, Guidelines for 10CFR50.59 Implementation, November 2000.

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[1] The printed name, company, department, and date must be included on the form. Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

5. NRC License Amendment 220, Waterford 3, Modification of Technical Specification 3/4.9.6 Refueling Machine, June 4, 2009.
6. NRC License Amendment 235, Waterford 3, Request to Revise the Technical Specifications Based Upon a Revised Fuel Handling Accident Analysis, April 25, 2012.
7. NRC License Amendment 243, Waterford 3, Relocation of Technical Specification 3/4.9.6 and 3/4.9.7 to the Technical Requirements Manual, July 29, 2015.
8. CN-NFPE-09-57 Revision 0, Waterford and Arkansas Major Fuel Handling Accident Evaluation, June 24, 2009.
9. EC16302, FUEL HANDLING ACCIDENT (FHA) ALTERNATIVE SOURCE TERM (AST) RADIOLOGICAL DOSE CONSEQUENCES FOR 3716 MWT EXTENDED POWER UPRATE (EPU), December 19, 2009.
10. EC81569 Technical Requirements Manual 3/4.9.6 Change.
11. EC81562 Refuel Machine CEA Beam

**Is the validity of this Evaluation dependent on any other change?** ☐ Yes ☒ No

**If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.**

**Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval?** ☐ Yes ☒ No

**Preparer[2]:** William Steelman / EC81569 for Signature / Entergy / Engineering / 2/10/19
Name (print) / Signature / Company / Department / Date

**Reviewer[2]:** James Hoss / EC81569 for Signature / Entergy / Engineering / 2/10/19
Name (print) / Signature / Company / Department / Date

**Independent Review[3]:** Not Required
Name (print) / Signature / Company / Department / Date

**Responsible Manager Concurrence:** BJ Parker / / Entergy / Refueling / 2/10/19
Name (print) / Signature / Company / Department / Date

**50.59 Program Coordinator Concurrence:** Remy Devoe / EC81569 for Signature / Entergy / Licensing / 2/10/19
Name (print) / Signature / Company / Department / Date

**OSRC:** Brian Lanka / EC81569 for Signature / Entergy / Engineering / 2/10/19
Chairman's Name (print) / Signature / Date [GGNS P-33633, P-34230, & P-34420; W3 P-151]
19-06
OSRC Meeting #

2 Either the Preparer or Reviewer will be a current Entergy employee.
3 If required by Section 5.1[3].

## II. 50.59 EVALUATION [10 CFR 50.59(c)(2)]

**Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.** ☐ Yes ☒ No

**Does the proposed Change:**

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

   BASIS:

   EC81569 will allow the refuel machine CEA beam to be used to safely lift a new fuel assembly that is not seated on the bottom core support plate alignment pins.

   The UFSAR was reviewed to identify which accidents previously evaluated could be initiated or caused by the proposed change. The plant is currently in Mode 6 refueling and the only accident impacted by this change is UFSAR Section 15.7.3.4 (Design Basis Fuel Handling Accidents). The UFSAR Section 15.7.3.4 fuel handling accident is defined as a limiting fault event. UFSAR Section 15.0.1 defines a limiting fault as incidents that are not expected to occur during the lifetime of the plant, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. UFSAR Section 15.7.3.4.1 describes the basis for the limiting faults as follows:

   > The possibility of a fuel handling accident is remote because of the many interlocks and administrative controls and physical limitations imposed on the fuel handling operations (refer to UFSAR section 9.1.4). All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety and fuel handling.

   The UFSAR Section 15.7.3.4.1 administrative controls, procedures, supervision, and training are all still consistent with the UFSAR information. The UFSAR Section 15.7.3.4 fuel handling accident analysis assumes that a fuel assembly is dropped. The fuel handling accident analysis [Reference 8] also assumes the limiting conditions such to maximize kinetic energy (height). For the fuel handling accident to occur, a irradiated assembly must be capable of being damaged. Fuel assembly LAHE20, is a new fuel assembly. For the identified change, the following 3 restrictions are required:

   1. Fuel assembly LAHE20 cannot be raised more than 6 inches off the bottom reactor core alignment plate [Reference 11]. This is controlled in the work order.
   2. Fuel assembly LAHE20 will always be positively captured until it is back in its normal configuration on the core alignment pins [Reference 10].
   3. A load cell will be monitored to prevent exceeding the overload limits [Reference 10].

   These restrictions provide administrative controls that provide similar restraints to the refuel machine interlock functions. The use of the manual lift rig and load cell is a slow process with limited risk of exceeding any of the limits. Since fuel assembly LAHE20 is a new fuel assembly, any damage to it would not be an accident because there is no possibility of a significant radioactivity release. In addition, by restricting the movement height, there is no possibility that fuel assembly LAHE20 could vertically drop and damage a irradiated fuel assembly because 6 inches is below the top of the fuel assemblies. Fuel assembly LAHE20 will remain positively captured until it is in its normal configuration, so this would prevent a potential tipping scenario and eliminates that from the frequency of occurrence. This means the event frequency category would not be impacted. Therefore, the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

NEI 96-07 [Reference 4] Section 4.3.2 states that the determination of whether the likelihood of malfunction is more than minimally increased is made at a level consistent with existing UFSAR described failure modes and effects analyses. UFSAR Section 9.1.4.3.2 (Fuel Handling) states the following:

> A failure mode analysis is not required.
> The results of the safety analysis (Chapter 15) demonstrate that applicable dose limits (10CFR50.67) are not exceeded as a result of the design basis fuel handling accident. No credit is taken for components or subsystems of the fuel handling equipment to either prevent or mitigate the consequences of the postulated accident.

The UFSAR Section 9.1.4.3.2 information would indicate no increase in the likelihood of malfunction of the refuel machine with respect to fuel movement, but the evaluation must also consider the nature of the proposed change. The term "malfunction of an SSC important to safety" refers to the failure of structures, systems and components (SSCs) to perform their intended design functions. UFSAR Section 9.1.4 describes that the fuel handling equipment includes interlocks, travel limiting features, and other protective devices to minimize the possibility of inadvertent damage to a fuel assembly and potential fission product release, resulting from either mishandling or equipment malfunction. UFSAR Section 9.1.4.2.1.1 lists the interlocks associated with the refuel machine. Technical Requirements Manual 3/4.9.6 requires an overload cut off limit.

This change will be using equipment that does not contain the refuel machine interlocks. This change is specifically due to a new fuel assembly LAHE20 not being seated on the bottom core plate alignment pins. NRC license amendment 220 [Reference 5] previously addressed a similar issue to place a fuel assembly in a safe condition while the refuel machine was inoperable. NRC license amendment 220 states the following:

> The proposed amendment to TS 3/4.9.6 clarifies previous wording that hindered the refueling machines operator's ability to recover from an undesirable situation. The change in wording allows the refueling machine operator to manually perform the function of the interlocks, normally controlled by the refueling machine computer, in the event that the computer fails mid-hoist. In this case, the operator action would meet the guidance of SRP 9.1.4 and GDC 61, which limit the load being hoisted and limit load movement in order to prevent fuel damage or radioactivity release. There, the NRC staff finds the proposed change acceptable.

While this change is not for the same condition as described in the NRC license amendment 220, it does demonstrate the NRC has previously approved a similar situation which allows the operator to perform the function of the refuel machine interlocks when it is necessary to place the fuel in a safe condition. In addition, UFSAR Section 9.1.4.2.2.5 (Fuel Handling

Tools) states that:

> Two fuel handling tools, as shown on UFSAR Figure 9.1-7, are used to move fuel assemblies in the spent fuel pool area. A short tool is provided for dry transfer of new fuel, and a long tool is provided for underwater handling of both spent and new fuel in the spent fuel pool. The tools are operated manually.

In terms of increasing the likelihood of a malfunction, the manual operation of refueling tools in the spent pool area would be similar to the manual operating of refuel tools in the reactor cavity.

NEI Section 4.3.2 states that qualitative engineering judgment and/or an industry precedent is typically used to determine if there is more than a minimal increase in the likelihood of occurrence of a malfunction. The refuel machine design function is to move fuel and prevent the fuel from being damaged. This change will allow the refuel machine CEA beam to be used to lift a fuel assembly that is not seated on the bottom core plate alignment pins. The use of the CEA beam, load cell, and manual rigging will take the place of the refuel machine interlocks. These interlocks are not credited in the UFSAR Section 15.7.3.4 fuel handling accident for accident mitigation and the operator action is not required to support a design function credited in the safety analysis (NEI Section 4.3.2 Example 4). The equipment is designed to the rated requirements and indication will be used to ensure overload conditions do not occur. Based upon the information discussed, there is no qualitative increase in the likelihood of occurrence of a malfunction. From a precedent perspective, the NRC license amendment 220 previously allowed placing the fuel in a safe condition and crediting an operator to perform the interlock function as acceptable. Thus, there is not a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators. The plant is currently in Mode 6 refueling and the only accident impacted by this change is UFSAR Section 15.7.3.4 (Design Basis Fuel Handling Accidents).

The limiting consequences of the UFSAR Section 15.7.3.4 fuel handling accident are the result of a irradiated assembly dropping vertically and impacting another irradiated assembly which results in all fuel pins failing [Reference 8 and 9]. This activity remains bounded by the UFSAR Section 15.7.3.4 fuel handling accident because:

- The temporary rig design and work order controls prohibit up movement of the new fuel assembly LAHE20 greater than 6 inches above the core alignment plate. This means that fuel assembly LAHE20 cannot impact another fuel assembly from the vertical direction.
- Fuel assembly LAHE20 is a new fuel assembly that has not been irradiated by an operating reactor core. Thus, the radioactive gases in the gap region of the fuel rod are much less than those assumed in the accident analysis.
- Fuel assembly LAHE20 will be positively secured during the entire evolution,

thus preventing the possibility of tipping over.

- The limiting UFSAR Section 15.7.3.4 analysis fails all the pins in the falling and impacted fuel assemblies. The analysis assumes both assemblies have been irradiated by an operating reactor. This activity involves a new fuel assembly so there is no possibility of failing 2 irradiated assemblies.

In summary, the radiological consequences of a fuel handling accident as described in UFSAR 15.7.3.4 are not increased and remain within the 10CFR50.67 limits.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?

☐ Yes
☒ No

BASIS:

When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators. The plant is currently in Mode 6 refueling and the only accident impacted by this change is UFSAR Section 15.7.3.4 (Design Basis Fuel Handling Accidents).

UFSAR Section 9.1.4.3.2 states the following:

> The results of the safety analysis (Chapter 15) demonstrate that applicable dose limits are not exceeded as a result of the design basis fuel handling accident. No credit is taken for components or subsystems of the fuel handling equipment to either prevent or mitigate the consequences of the postulated accident.

Since the fuel handling equipment does not prevent or mitigate the consequences of an accident, this change cannot increase the consequences of a malfunction.

5. Create a possibility for an accident of a different type than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

This change will allow the refuel machine CEA beam to be used to lift a fuel assembly that is not seated on the bottom core plate alignment pins. The use of the CEA beam, load cell, and manual rigging will take the place of the refuel machine interlocks. These interlocks are not credited for accident mitigation in the UFSAR Section 15.7.3.4 fuel handling accident and the operator action is not required to support a design function credited in the safety analysis (NEI 96-07 Section 4.3.2 Example 4). The equipment is designed to the rated requirements and indication (load cell) will be used to ensure overload conditions do not occur.

UFSAR Chapter 6 and 15 were reviewed to identify types of accidents that might be different than those previously evaluated. The fuel handling accident is already described in the UFSAR Section 15.7.3.4 (Design Basis Fuel Handling Accidents). The proposed change is only related to moving fuel and the current UFSAR addresses the potential accidents associated with moving fuel.

Based upon the limiting potential impacts, no creditable accident of a different type can be postulated.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:

This activity could result in new fuel assembly LAHE20 being damaged and/or dropped during the evolution. However, this is not a malfunction with a different result since the effect of this malfunction is bounded by the existing UFSAR Section 15.7.3.4 fuel handling accident analysis which determined that two fuel assemblies have all their fuel pins fail. Thus, there is not a possibility for a malfunction with a different result than previously evaluated in the UFSAR.

7. Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

The fission product barriers are the fuel cladding, reactor coolant system (RCS) pressure boundary, and containment. Existing design basis limits for the fuel, RCS pressure boundary, and containment are not altered as a result of implementing the proposed change. The temporary hoist design and established procedural controls preclude damage to or dropping of new fuel assembly LAHE20 during movement such that fuel assembly design limits will continue to be met. Any fuel handling issue would continue to be bounded by the existing UFSAR Section 15.7.3.4 analysis which determined that two fuel assemblies have all their fuel pins fail. Further, there is no change or impact to systems or components that are credited to mitigate design basis accidents and preserve fission product barriers. Based on this, there is no impact to or compromise of the existing fission product barrier limits.

8. Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

Method of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC [Reference 4 Section 3.10]. This change is not impacting the methodology or topical reports used in the UFSAR analyses. Therefore, the proposed change does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

**If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.**

## I. OVERVIEW / SIGNATURES[1]

**Facility:** Waterford 3 **Evaluation # / Rev. #:** 2019-03 / 0

**Proposed Change / Document:** EC 82583 Calculation Updates Addressing CR-WF3-2018-7188 and 7253

**Description of Change:**
The scope of this 50.59 evaluation is limited to the portion of the change involving the addition of securing the AH-2A(C) fan units via associated breakers to the existing action to secure the Safeguards A Pump room train A pumps following a loss of coolant accident (LOCA), loss of Essential Chiller A, without a loss of offsite power (LOOP), and High Pressure Safety Injection (HPSI) pump AB aligned to train B. The failure of Essential Chiller A is bounding because it causes both AH-2A and C to provide no cooling. Failure of subcomponents such as valves in the system could result in only challenging one of the two coolers, which is less limiting. Larger failures such as that of an emergency diesel generator (EDG) would result in the pumps and fans securing which would remove the heat loads and eliminate the need for the new operator action of securing the fans in addition to the existing operator action of securing the safeguards A pumps.

The current operator action in annunciator response procedure OP-500-013 Attachment 4.29 Safeguards Pumps A Area Temperature Hi only requires securing the safeguards A pumps in the room following a LOCA and single failure of Essential Chiller A. However, the AH-2A(C) fans need to be secured to reduce the heat load further in order to be within the capacity of the remaining AH-21 cooler per calculations ECM97-031 and 5-A. This action must be performed within 8.5 hours following receipt of the high temperature annunciator (CP18-64, via HVR-ITAC-5003A) per annunciator response procedure OP-500-013 and supporting calculations ECM97-031 and 5-A. The annunciator takes up to 30 minutes to occur post-accident so the total time from the onset of the event is 9 hours to perform the action, 8.5 hours after indication is received. The existing operator action being modified by this change is also being added to the Time Critical Action program (EN-OP-123) as a part of this change.

Based on a review of the control wiring diagrams B424 Sheets 1005 and 1006, the AH-2A(C) fan unit(s) will automatically secure when the associated train A pumps in the room are secured unless the local control switch was used to start the unit(s) and a LOOP has not occurred. Using the local control switch to start the AH-2A(C) units results in the ONX relay creating a lock-in circuit such that the fans will not secure when the associated train A safeguards pumps are secured and the 52 relays open. If a LOOP occurs, the ONX relay will de-energize, resetting the relays in the circuit to their shelf-state, which clears the lock-in circuit. This operator action is therefore only required if the AH-2A(C) units were locally started via the local control switch prior to the LOCA and a LOOP has not occurred.

The remaining changes in EC 82583 are addressed in the associated Process Applicability Determination per EN-LI-100.

**Summary of Evaluation:**
The following 50.59 evaluation shows that the addition of a requirement to secure the AH-2A(C) fan units to the existing requirement of securing the train A safeguards pumps in the Safeguards A Pump room following a LOCA and loss of Essential Chiller A without a LOOP may be implemented without prior NRC approval.

[1] The printed name, company, department, and date must be included on the form. Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

The Safeguards A Pump room contains the following pumps: Containment Spray (CS) Pump A, Low Pressure Safety Injection (LPSI) Pump A, and both High Pressure Safety Injection (HPSI) Pump A and AB. The HPSI AB pump has its own fan cooler, AH-21, which is serviced by Essential Chiller B. The remaining train A pumps in the room are cooled by the AH-2A(C) fan coolers, which are serviced by Essential Chiller A.

When Essential Chiller A is lost, the AH-2A(C) fans and associated motors only add heat to the room since the associated cooling coils served by Essential Chiller A are providing no cooling. Other failures in the Chilled Water System such as that of chilled water pumps, isolation valves, or cooling coils at a specific cooler are bounded by the Essential Chiller failure. The added heat load of these fans in the post-Recirculation Actuation Signal (RAS) room environment would result in exceeding the heat removal capacity of the one remaining room cooler in the Safeguards A Pump room, AH-21 which is served by Essential Chiller B when the HPSI AB pump is aligned to train B, as is assumed in this limiting case. Therefore annunciator response procedure OP-500-013 is required to be updated to include securing the AH-2A(C) fan units via their associated breakers in the Switchgear A room if they do not automatically secure when the train A pumps are secured. This change preserves the safety functions of the HPSI AB pump in the Safeguards A Pump room.

**Is the validity of this Evaluation dependent on any other change?** ☐ **Yes** ☒ **No**

**If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.**

**Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval?** ☐ **Yes** ☒ **No**

**Preparer[2]:** (For qual) Alex Tojeiro / [signature] / EOI / DESMECH / 6/20/19

William Steelman / [signature] EOI / Eng / 7-10-19

Name (print) / Signature / Company / Department / Date

**Reviewer[2]:** James W. Hoss [signature] EOI / Eng / 7-10-19

Name (print) / Signature / Company / Department / Date

**Independent Review[3]:** NA

Name (print) / Signature / Company / Department / Date

**Responsible Manager Concurrence:** Nicholas Petit / [signature] / EOI / ENG / 7-10-19

Name (print) / Signature / Company / Department / Date

**50.59 Program Coordinator Concurrence:** Remy DeVoe / [signature] / EOI / RA / 7-10-19

Name (print) / Signature / Company / Department / Date

**OSRC:** RAN B. GILMORE / R.B. Gilm / 7-10-19 /

Chairman's Name (print) / Signature / Date [GGNS P-33633, P-34230, & P-34420; W3 P-151]

[2] Either the Preparer or Reviewer will be a current Entergy employee.
[3] If required by Section 5.1[3].

19-14
OSRC Meeting #

## II. 50.59 EVALUATION [10 CFR 50.59(c)(2)]

**Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.** ☐ Yes ☒ No

**Does the proposed Change**:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

   BASIS:
   The modified operator action evaluated herein is the addition of securing the AH-2A(C) fan units via their associated breakers to the existing action to secure the Safeguards A Pump room train A pumps following a LOCA, loss of Essential Chiller A without a LOOP, and HPSI pump AB aligned to train B. This action is only required if the fan units were started via the local control switch prior to the LOCA. The proposed change is limited to a LOCA where a RAS occurs. Other accident scenarios are not involved in this change since the operator action associated with this change is to limit overheating in the room after RAS has occurred, where the piping in the room is at the temperature of the Safety Injection Sump (SIS) rather than the Refueling Water Storage Pool (RWSP) and therefore creates a heat load rather than a heat sink.

   LOCA events in Chapters 6 and 15 of the UFSAR were reviewed. The operator action of securing the AH-2A(C) fan units occurs after a LOCA has already occurred and only if Essential Chiller A is lost. Securing the AH-2A(C) fan units cannot cause an accident to occur since failure of the units is not a potential accident initiator. The loss of room cooling to the Safeguards A pump room during normal operation does not cause an accident to occur. Therefore the proposed change has zero impact on the frequency of occurrence of an accident previously evaluated in the UFSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

   BASIS:
   The modified operator action evaluated herein is the addition of securing the AH-2A(C) fan units via their associated breakers to the existing action to secure the Safeguards A Pump room train A pumps following a LOCA, loss of Essential Chiller A without a LOOP, and HPSI pump AB aligned to train B. This action is only required if the fan units were started via the local control switch prior to the LOCA. This change involves a modified operator action that supports a design function credited in the safety analyses. This is reviewed against the four items listed in NEI 96-07 Revision 1, Section 4.3.2 Example 4:

   1. The action (including required completion time) is reflected in plant procedures and operator training programs

      As a part of this change the existing guidance in operating procedure OP-500-013 is updated to include securing the AH-2A(C) fans via their associated breakers HVR-EBKR-313A-4K and HVR-EBKR-313A-5K respectively (tracked by AR-19005503 AS-0300). This procedure currently includes the applicable time limit of 8.5 hours after the alarm (CP18-64) is received, which occurs within approximately 30 minutes post-accident. This timeline for operator response remains applicable for this modified action. The action may be performed at any time prior to the design basis 9 hour limit from onset of the accident.

This action is added to the Time Critical Action program (tracked by AR-19005503 AS-0200) such that it requires validation. This program is based on PWROG-16030-NP, Time Critical Action/Time Sensitive Action Program Standard. The updated guidance in OP-500-013 is sufficient for crediting this action. The training and validation requirements of the Time Critical Action program will reinforce this training at a later date and are not required for crediting the action.

2. The licensee has demonstrated that the action can be completed in the time required considering the aggregate affects, such as workload or environmental conditions, expected to exist when the action is required.

This action is specific to the case where the HPSI AB pump is aligned to train B and the limiting single failure is the train A Essential Chiller. The AH-2A(C) fans cannot be remotely secured from the control room, however the Safeguards A pumps can. The post-RAS environmental conditions of the room will involve high dose rates and high temperatures up to 152°F. Therefore, the change involves securing the AH-2A(C) fans by opening their associated breakers HVR-EBKR-313A-4K and HVR-EBKR-313A-5K. These breakers are located in the Switchgear A room. The breakers are located on MCC 3A313-S. The areas outside the Control Room in the potential travel paths have lower dose rates for the accident per radiation maps G-M0001, G-M0011, and G-M0013. Temperatures in the associated areas may reach a maximum temperature of 104°F per G-M0001, G-M0003, and G-M-0005, which due to the short duration required for the operator actions this temperature does not require taking any additional precautions for heat stress concerns.

EN-OP-123, “Time Critical Action/Time Sensitive Action Program Standard”, Attachment 10, “W3 Specific TCA Addendum” was reviewed to determine workload. Since the action time limit is 9 hours post-LOCA, Time Critical Actions (TCA) and Time Sensitive Actions (TSA) required within ±2 hours of this time were reviewed if applicable to a LOCA or loss of offsite power (LOOP) which is assumed to occur concurrently with the LOCA. Actions falling into this group are TCAs 23 and 24 which are required at 10 hours. TCAs 23 and 24 are in response to a loss of Instrument Air and involve a significant number of manual operator actions at various valves throughout the plant. TCA 23 drives operating the manual backup Essential Air regulator isolation valves in the RAB and Switchgear Wing Areas. TCA 24 involves:

- Taking manual control of EFW-223A(B), EFW-224A(B), ACC-126A(B), and MS-116A(B), and gagging the following
- Gagging the following valves:
    - CC-114A(B), CC-115A(B), CC-126A(B), CC-127A(B) (one set of valves)
    - CC-200A(B), CC-727, CC-563 (one set) as needed to maintain CCW AB header flow and SFP cooling.

The above actions for addressing the loss of instrument air and/or nitrogen capacity require significant operator action and may be challenged if they are not started well in advance of the associated 10 hour limit. However, the closest preceding action is at 6 hours such that there is a 4 hour period to perform the above TCAs 23 and 24, in addition to the manual actions for the Safeguards A Room pumps and fans. This provides ample time to complete the actions such that the addition of needing to secure the fans will have a negligible impact on the operators’ workload during this phase of the accident. In addition, since the evaluated action is only required in non-LOOP events the non-safety related Instrument Air system would be expected to be available such that TCAs 23 and 24 may not be necessary during the event where the evaluated action is required.

Using ANSI/ANS-58.8-1994 “Time Response Design Criteria for Safety-Related Operator Actions”, the total time required to perform this action can be calculated as follows:

$TI_{ind}$: 30 minutes. This is the time required for the high temperature alarm/annunciator to occur post-LOCA.
$T_{\cdot diagnosis}$: 20 minutes. Conservatively assumed based on a LOCA being Plant Condition 4 or 5.
$TI_{operator}$: 60 minutes. Per Table 2 of ANSI/ANS-58.8-1994, 30 minutes is a fixed minimum time required to be considered for this case, with one minute added per discrete manipulation required to complete the action. The 30 minutes accounts for necessary preparations to perform the action(s) and reaching the location the action is being performed (Switchgear). Opening the cabinet and the two breakers (one each for AH-2A and AH-2C) is conservatively assumed to take 10 minutes per discrete action rather than the standard one minute per action, yielding a total of 60 minutes for this stage.

For the purpose of this analysis, $TI_{process}$ for the process response is effectively instantaneous since it is the time it takes for the motor to de-energize following opening of its breaker. The time contribution of this step is considered negligible since it is on the order of fractions of a second versus the total time which is on the order of hours.

The action of opening the breakers completes the required safety function. Based on the conservative action time reviewed above, the total time required to complete the action will be 30 minutes + 20 minutes + 60 minutes = 110 minutes. Therefore, the action will be completed in less than two hours from the onset of the event. Given the allotted time of 9 hours from the onset of the event (8.5 hours following indication), the significant margin of over 300% demonstrates that validation of this operator action is not required prior to crediting the action in the design basis. However, validation will be required to be performed at a later time to formally incorporate the action in the Time Critical Action Program.

3. The evaluation of the change considers the ability to recover from credible errors in performance of manual actions and the expected time required to make such a recovery.

   The action in this evaluation is only required after the design basis single failure of Essential Chiller A has occurred such that subsequent failures do not need to be considered. For conservatism, the ability to recover from a subsequent failure of an operator to open the correct breaker is reviewed herein.

   After the required breakers are opened, the status of the associated AH-2A(C) fans can be verified by checking the indicator lights on Control Room panel CP-18 for each fan. Various computer points on the PMC provide indication of the status of the cooler (D43316) and differential pressure across the fans (D43306, D43308), and non-class 1E alarm and indication (B0201, D43311, and D43313) for the room coolers' power being lost provide sufficient redundancy in indication such that if the wrong breaker is opened, the error will be able to be quickly identified by a lack of status change in the described indications.

4. The evaluation considers the effect of the change on plant systems.

   Securing the AH-2A(C) fans by opening their associated breakers HVR-EBKR-313A-4K and HVR-EBKR-313A-5K only impacts the operation of the fans. There is no additional equipment downstream of the breakers that would be tripped by opening the breakers. Therefore, there is no potential adverse effect on plant systems as a result of the modified operator action.

The above discussion also addresses concerns raised in Information Notice 97-78, with the exception of risk significance of the change. Due to the extremely long available time for performing this action, the available margin, and available indication for taking appropriate action, the impact of this change on plant risk is considered negligible. The risk significance of the fans in the Safeguards A room has not changed as a result of the modified operator action.

Based on the preceding discussion, the modified operator action of including the AH-2A(C) fans as being required to be secured via their breakers to the existing operator action to secure the Safeguards A Pumps following receipt of a high room temperature alarm per operations procedure OP-500-013 has a less than minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators.

The modified operator action evaluated herein is the addition of securing the AH-2A(C) fan units via their associated breakers to the existing action to secure the Safeguards A Pump room train A pumps following a LOCA, loss of Essential Chiller A without a LOOP, and HPSI pump AB aligned to train B. This action is only required if the fan units were started via the local control switch prior to the LOCA. The proposed change is limited to a LOCA where a RAS occurs (i.e. small break and large break LOCA). Other accident scenarios are not involved in this change since the operator action associated with this change is to limit overheating in the room after RAS has occurred, where the piping in the room is at the temperature of the Safety Injection Sump (SIS) rather than the Refueling Water Storage Pool (RWSP) and therefore creates a heat load rather than a heat sink.

The AH-2A(C) breakers (HVR-EBKR-313A-4K and HVR-EBKR-313A-5K) are located in the Switchgear A room, which is accessed from the Control Room by taking the stairs northeast from the Control Room down two levels to the +21 elevation, then heading south into the Switchgear A room. The breakers are located on MCC 3A313-S.

The accident dose for a small break LOCA is described in calculation ECS04-013, and for a large break LOCA is ECS04-001. The calculated control room operator dose for each case is 3.366 rem and 3.385 rem respectively. The areas outside the Control Room in the travel path do not have elevated dose rates during an accident per radiation maps G-M0001, G-M0011, and G-M0013. The radiation maps are based on the bounding maximum dose rates in the associated areas for any accident. The path from the control room would result in higher dose than if the operator relocates to the switchgear from outside the nuclear plant island structure (NPIS), because dose rates in the control room are elevated post-accident and the switchgear areas (and surrounding hallways) do not experience elevated dose rates post-accident. Therefore, the travel path required to be taken to complete this action results in less dose than if the operator were to remain in the control room such that the resultant dose to any operator does not exceed the calculated dose in ECS04-013 and ECS04-001, or the General Design Criterion 19 limit of 5 rem whole body (or its equivalent to any part of the body) for the duration of the applicable accidents. If the operator began outside the control room, the evaluated control room dose for operators would not be exceeded for performing this action.

Securing the fan units results in protecting the remaining single train of HPSI (the AB pump) in operation and results in the required safety functions of the Safety Injection system to be continued to be met. This manual action does not introduce any additional release paths or adversely affect dose rates to control room personnel or the public as calculated by ECS04-013 and ECS04-001.

Since the travel path does not introduce additional dose to the operators inside the control room or the public, there is zero increase in the consequences of an accident previously evaluated in the UFSAR as a result of this change.

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4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators.

The modified operator action evaluated herein is the addition of securing the AH-2A(C) fan units via their associated breakers to the existing action to secure the Safeguards A Pump room train A pumps following a LOCA, loss of Essential Chiller A without a LOOP, and HPSI pump AB aligned to train B. This action is only required if the fan units were started via the local control switch prior to the LOCA.

The failure modes for the Essential Services Chilled Water System provided in UFSAR Table 9.2-17 were reviewed. There is no failure modes table provided in the UFSAR for the Safeguards Room coolers. The failures of the Essential Chiller credit the opposite train for mitigation. There are no dose consequences for postulated limiting failures of an Essential Chiller.

This action is taken after the AH-2A(C) has already lost the capability to perform its safety function due to the postulated single failure of Essential Chiller A. The action is taken to reduce the heat load in the Safeguards A room in order to preserve the ability of the HPSI AB pump to perform its safety functions. Securing the fan units does not introduce dose consequences to the postulated malfunction of Essential Chiller A. This change does not increase reliance on any structures, systems, or components. Therefore, there is zero increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
The modified operator action evaluated herein is the addition of securing the AH-2A(C) fan units via their associated breakers to the existing action to secure the Safeguards A Pump room train A pumps following a LOCA, loss of Essential Chiller A without a LOOP, and HPSI pump AB aligned to train B. This action is only required if the fan units were started via the local control switch prior to the LOCA. This action occurs after a LOCA and does not have the potential to initiate any type of accident. As discussed in NEI96-07 Section 4.3.5, accidents that would require multiple independent failures or other circumstances in order to be created would not meet this criterion. The proposed modified operator action is only applicable after a specific accident has occurred with a single failure of a specific component.

Since this action is performed after the accident and single failure have occurred, and in itself cannot initiate another accident, this modified operator action cannot create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR? ☐ Yes ☒ No

BASIS:
The modified operator action evaluated herein is the addition of securing the AH-2A(C) fan units via their associated breakers to the existing action to secure the Safeguards A Pump room train A pumps following a LOCA, loss of Essential Chiller A without a LOOP, and HPSI pump AB aligned to train B. This action is only required if the fan units were started via the local control switch prior to the LOCA.

Malfunctions of SSCs are generally postulated as potential single failures to evaluate plant performance with the focus being on the result of the malfunction rather than the cause or type of malfunction. A malfunction that involves an initiator or failure whose effects are not bounded by those explicitly described in the UFSAR is a malfunction with a different result. The malfunction applicable to this question is determining whether securing the AH-2A(C) fan units creates a different result that is not bounded by previously evaluated failures of Essential Chiller A. The AH-2A(C) fan units are designed to automatically secure when all the associated train A Safeguards Pumps are secured if the local control switch is not used. Since the fans are already designed to be secured following the pumps being secured, performing this action manually by opening the breakers does not introduce a malfunction with a different result. Securing the AH-2A(C) fan units does not adversely affect Essential Chiller heat loads on the remaining operating train. Securing the AH-2A(C) fan units preserves the ability of the remaining operating HPSI AB pump in the Safeguards A Room to perform its associated safety functions by reducing the heat load in the room to be within the capacity of the remaining room cooler (AH-21) such that the malfunction remains bounded by the existing failure modes analysis in UFSAR Table UFSAR Table 9.2-17 for the Essential Chillers. There are no adverse impacts to other systems resulting from securing the AH-2A(C) fan units. From a high level perspective, Essential Chiller B will remain capable of meeting the safety function of the Essential Chilled Water system such that there is no loss of safety function as a result of this operator action. Therefore, the proposed modified operator action does not create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR.

7. Result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:
The modified operator action evaluated herein is the addition of securing the AH-2A(C) fan units via their associated breakers to the existing action to secure the Safeguards A Pump room train A pumps following a LOCA, loss of Essential Chiller A without a LOOP, and HPSI pump AB aligned to train B. This action is only required if the fan units were started via the local control switch prior to the LOCA.

The AH-2A(C) fan coolers do not directly protect any fission product barriers. Since the fan coolers normally support operation of the train A safeguards pumps, which includes the HPSI, LPSI and CS pumps, the safety functions of the Safety Injection and Containment Spray systems are dependent on the ability of these room coolers to perform their safety function. In the specific limiting failure where the subject operator action would occur, Essential Chiller A is the postulated single failure such that the AH-2A(C) is incapable of performing its safety functions since it is dependent on Essential Chiller A to provide heat removal to the space. Securing the fans as intended by the modified operator action ensures the remaining HPSI AB pump in the Safeguards A Room remains functional.

Since in this event the train B CS and LPSI pumps and the HPSI AB pump (in place of HPSI B pump) would be unaffected by both the failure of Essential Chiller A and the action of securing the AH-2A(C) fan units, the safety functions of the safeguards systems would remain met, and the remaining fission product barriers of the fuel cladding and containment would not be challenged (RCS lost due to initiating event of a LOCA). Therefore, the proposed modified operator action does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:
"Methods of evaluation" means the calculational framework used for evaluating behavior or response of the facility or an SSC. As discussed in the associated 50.59 screening, the proposed operator action does not involve or affect any methods of evaluation described in the UFSAR. Therefore, the proposed change does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

**If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.**

# I. OVERVIEW / SIGNATURES[1]

**Facility:** Waterford 3 **Evaluation # / Rev. #:** 20-001 Rev 0

**Proposed Change / Document:** EC-86397, MPC #561 Lid Helium Drain/Vent Cover Plate Repair

**Description of Change:** During closure operations for MPC-32 S/N 561 at Waterford, personnel were unable to complete the plug weld over one of the 1/4-20UNC set screws (Item 20 on DWG 3753) in the vent port cover plate due to helium pressure buildup beneath the cover plate (ref. CR-WF3- 2020-1455). It is proposed to cut out both the vent and port cover plates, re-perform FHD drying and helium backfill operations, and proceed with closure operations using new Alloy X cover plates (see IPR-2849-112-R0) that utilize stainless steel 1/8" NPT threaded plugs in place of the original port cover plates and set screws. All material thicknesses of the new cover plates are identical to those of the original plate design. The NPT threaded plugs are fully threaded into the cover plate and plug welded in place, thereby performing an identical function as the set screws in the original cover plate design. All sealing operations, including nondestructive inspection and leak testing to demonstrate compliance with the HI-STORM 100 FSAR & CoC, shall be completed in accordance with Waterford Procedure DFS-003-005.

See SMDR-1023-2895 for additional details of the proposed replacement.

It should be noted that while this condition is currently applicable to MPC S/N 561, the new port cover (both Vent and Drain) can be used for any future MPC.

Minor helium leakage beneath the port cover plates is not a concern as long as all closure welds on the cover plate can be completed to establish the MPC confinement boundary and subsequent leak testing confirms the confinement boundary is leaktight in accordance with the CoC. Note that though the port cover plates are part of the MPC confinement boundary, the port tubes and port caps are not and thus, are not required to be leaktight.

As discussed in Attachment A of SMDR-1023-2895, the weld chamfer around the circumference of the new port cover plates is identical to that of the original cover plates, so the strength of the resulting welds is unchanged. In addition, the depth and diameter of the plug welds securing the NPT threaded plugs into the new cover plates are greater than those implemented on the original cover plates, so the strength of the plug welds is slightly increased. Though the 1/8” NPT threaded plugs require larger counterbores (Ø1/2” x 0.14"↓) in the port cover plates than the original set screw configuration (Ø1/4”-20UNC recessed ≥1/8”), all penetrations will be completely filled with weld metal and the material thicknesses for the new cover plates are identical to those specified for the original cover plates, so the structural integrity of the plates themselves are maintained. All material characteristics remain consistent with the licensing basis and all welds will be inspected and tested in accordance with the CoC and FSAR requirements, so the structural analyses provided in Supplement 40 of HI-2012787 remain applicable. With the cover plate welding completed, leakage past the port tube and/or port cap is contained and will stop once the port cavity beneath the cover plate reaches equilibrium pressure with the full MPC volume. The port tube and port cap are not credited as part of the MPC confinement boundary, so the confinement design function is met by the welded port cover plate and plug welded set screws/threaded plugs. As such, the confinement design function of the MPC will be maintained.

Because all material thicknesses for the new port cover plates are identical to those of the original plates, the shielding contribution from the MPC is preserved, and the HI-TRAC and HI-STORM are not affected by the proposed activity, so the overall shielding performance of the system is not affected.

[1] The printed name should be included on the form when using electronic means for signature or if the handwritten signature is illegible. Signatures may be obtained via electronic authentication, manual methods (e.g., ink signature), e-mail, or telecommunication. Signing documents with indication to look at another system for signatures is not acceptable such as “See EC” or "See Asset Suite." Electronic signatures from other systems are only allowed if they are included with the documentation being submitted for capture in eB (e.g., if using an e-mail, attach it to this form; if using Asset Suite, attach a screenshot of the electronic signature(s); if using PCRS, attach a copy of the completed corrective action).

The proposed activity does not affect the thermal or criticality control design functions of the MPC-32 or the HI-STORM 100 system, nor does it significantly affect fabrication. The new port cover plates facilitate MPC closure operations and no other aspect of the loading operations are impacted. Leakage past the port tube and/or port cap will result in the cavity beneath the port cover plate reaching an equilibrium pressure with the rest of the MPC cavity during storage; however, the volume of this port cavity is small enough that the one-time release during unloading operations will not cause any regulatory requirement to be violated. The ability to complete loading operations with minor leakage past the port tube and/or port cap (prior to closure welding of the cover plate and NPT plugs) provides reasonable assurance that unloading operations can be similarly completed and that all regulatory requirements will continue to be met.

**Summary of Evaluation:** Minor helium leakage around the vent port tube is not a concern as long as all closure welds on the cover plate can be completed to establish the MPC confinement boundary and subsequent leak testing confirms the confinement boundary is leaktight in accordance with the HOLTEC CoC, provided helium backfill pressure is also demonstrated to remain compliant with the Technical Specifications.

To address the Technical Specification pressure limits as it related to the current leak rate past the port cap, after greater than 72 hours of leakage, when the vent port was recovered and MPC internal pressure was measured, the MPC pressure was still in compliance with the Technical Specification limits. The actual time to weld and close the port cover plate is less than 3 hours, thus is reasonable to assume that after the welding of the port cover plate, we will still be in compliance with the Technical Specification pressure limits despite the leak rate.
.
Note that though the vent port cover plate is part of the MPC confinement boundary, the vent port cap is not and thus, it is not required to be leaktight as discussed in Section 7.0 of the HI-STORM 100 CFSAR. As discussed in Attachment A of SMDR-1023-2895, the weld chamfer around the circumference of the new vent port cover plate is identical to that of the original cover plate, so the strength of the resulting weld is unchanged. In addition, the depth and diameter of the plug welds securing the threaded plugs into the new cover plate are greater than those implemented on the original cover plate, so the strength of the plug welds is slightly increased. Though the 1/8” NPT threaded plugs require larger bores (07/16” x 1/4”) in the port cover plate than the original set screw configuration (01/4”-2OUNC recessed 1/8”), all penetrations will be completely filled with weld metal and the overall material thickness of the new cover plate meets or exceeds the minimum thickness specified for the original cover plate, so the structural integrity of the plate itself is maintained. All material characteristics remain consistent with the licensing basis and all welds are inspected and tested in accordance with the HOLTEC CoC and CFSAR requirements, so the structural analyses provided in Supplement 40 of the HOLTEC analysis HI-2012787 remain applicable. With the cover plate welding completed, leakage from the MPC vent port tube is contained and will stop once the vent port cavity beneath the cover plate reaches equilibrium pressure with the full MPC volume.

The MPC vent port tube and threaded cap are not credited as part of the MPC confinement boundary, as discussed in Section 7.0 of the HI-STORM 100 CFSAR, so this function is met by the welded port cover plate and plug welded set screws/threaded plugs. As such, the confinement design function of the MPC will be maintained.

The proposed activity does not affect the thermal or criticality control design functions of the MPC-32 or the HI-STORM 100 system, nor does it significantly affect fabrication. The new port cover plate facilitates MPC closure operations and no other aspect of the loading operations are impacted. Leakage around the vent port tube will result in the cavity between the vent port cap and vent port cover plate reaching an equilibrium pressure with the rest of the MPC cavity during storage; however, the volume of the port cavity is small enough (—320 cc) that the one-time release during unloading operations, which is controlled in accordance with Chapter 8, Section 8.3.3 of the HI-STORM 100 CFSAR and Radiation Protection procedures, will not cause any regulatory requirement as identified in NUREG-1536 to be violated.

Addressing the potential loss of helium through the port cap into the area below the port cover, EC-32873, *DOCUMENT HELIUM BACKFILL NUMBERS FOR HOLTEC MPC-32 AT WATERFORD 3* calculated the MPC

Net Free Volume $V_{MPCNFV}$ = 248.9 $ft^3$ or 30,099.2 $in^3$ and the volume below the port cover approximately 320 $cm^3$ or 19.53 $in^3$. With the MPC pressure meeting the Cask Tech Spec 3.3.1, Table 3-2 MPC Helium Backfill Limits1, the MPC pressure should be at a minimum of 29.3 psig (reference Temperature at 70 degrees F). Thus, the pressure past the leaking port cap will come to equilibrium with the MPC itself and based on the extremally small volume beneath the port cover, there will have no significant impact on the MPC pressure.

Holtec has provided 72.48 Screening/Evaluation No. 1448 Rev.0. Entergy Procedure EN-LI-115 Rev. 7, HI-STORM 100 Independent Spent Fuel Storage Installation Licensing Document Preparation and Control, Section 5.3.1 contains the following note:

"*A 72.48 Screening or Evaluation record that is produced by the Certificate Holder can be used to satisfy Entergy's 72.48 review requirement."*

**Is the validity of this Evaluation dependent on any other change?** ☐ **Yes** ☒ **No**

**If "Yes," list the required changes/submittals. The changes covered by this 72.48 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.**

**Based on the results of this 72.48 Evaluation, does the proposed change require prior NRC approval?** ☐ **Yes** ☒ **No**

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**Preparer:** Jason Laque / signature in IAS Matrix/Projects/ 4/2/2020 /
Name (print) / Signature / Company / Department / Date

**Reviewer:** Gregory Ferguson, / signature in IAS Enercon/Engineering 4/2/2020 /
Name (print) / Signature / Company / Department / Date

**OSRC:** Brian Lindsey / Approval per email in p2E 4/2/2020
Chairman's Name (print) / Signature / Date **[QAPM A.2.f]**

20-03
OSRC Meeting #

## II. 72.48 EVALUATION

**Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.**

☐ Yes
☒ No

**Does the proposed Change**:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the CFSAR?

   ☐ Yes
   ☒ No

   **BASIS:**

   THE ACCIDENTS EVALUATED IN CHAPTER 11 OF THE HI-STORM 100 FSAR ARE ALL CAUSED BY NATURAL PHENOMENON (I.E. EXTREME TEMPERATURE, FLOOD ETC.) OR FORCES EXTERNAL TO THE HI-STORM (I.E. FIRE, HANDLING ACCIDENTS ETC.). THE CREDIBLE ACCIDENTS ARE POSTULATED TO OCCUR WITH A FREQUENCY OF 1.0 (THE MAXIMUM POSSIBLE FREQUENCY), THUS NO INCREASE IN FREQUENCY OF OCCURRENCE IS POSSIBLE FOR THESE EVENTS.

   LEAKAGE FROM THE CONFINEMENT BOUNDARY IS THE ONLY NON-CREDIBLE ACCIDENT EVENT THAT IS NOT SPECIFICALLY EVALUATED FOR THE HI-STORM 100 SYSTEM. THE EVENT IS DEEMED NON-CREDIBLE IN THE HI-STORM 100 FSAR BY CONFORMING TO THE REQUIREMENTS OF DIVISION OF SPENT FUEL STORAGE AND TRANSPORTATION INTERIM STAFF GUIDANCE - ISG-18, REV. 1, "THE DESIGN AND TESTING OF LID WELDS ON AUSTENITIC STAINLESS STEEL CANISTERS AS THE CONFINEMENT BOUNDARY FOR SPENT FUEL STORAGE". BECAUSE THERMAL PERFORMANCE OF THE MPC IS NOT AFFECTED, THE INTEGRITY OF THE CLOSURE WELDS IS NOT IMPACTED, AND AS DISCUSSED IN SECTION I, THE STRUCTURAL PERFORMANCE OF THE PORT COVER PLATES THEMSELVES ARE MAINTAINED, CONFINEMENT INTEGRITY IS MAINTAINED DURING ALL CONDITIONS. IN ADDITION, HELIUM LEAK TESTING OF THE PORT COVER PLATE WELDS AND BASE METAL WILL STILL BE PERFORMED TO DEMONSTRATE CONTINUED COMPLIANCE WITH THE LEAKTIGHT CRITERIA REQUIRED BY HI-STORM 100 COC CONDITION 3 AND LCO 3.1.1.

   THE REPLACEMENT OF THE PORT COVER PLATES (BOTH VENT OR DRAIN PORTS) WILL NOT IMPACT THE CONFINMENT BOUNDARY BECAUSE THE COVER PLATE AND FILLER MATERIAL ARE THE SAME FOR BOTH PLATES AND ARE BOUNDED BY CURRENT ANALYSIS.

   NOTE THAT THE PORT TUBE AND PORT CAP ARE NOT PART OF THE MPC CONFINEMENT BOUNDARY, SO THEY ARE NOT REQUIRED TO BE LEAKTIGHT AND ANY LEAKAGE AROUND THESE COMPONENTS DOES NOT CONSTITUTE A LEAKAGE OF THE CONFINEMENT BOUNDARY, PROVIDED THE INTEGRITY OF PORT COVER PLATE IS MAINTAINED. THROUGH CONTINUED COMPLIANCE WITH THE REQUIREMENTS OF DIVISION OF SPENT FUEL STORAGE AND TRANSPORTATION INTERIM STAFF GUIDANCE - ISG-18, REV. 1, "THE DESIGN AND TESTING OF LID WELDS ON AUSTENITIC STAINLESS STEEL CANISTERS AS THE CONFINEMENT BOUNDARY FOR SPENT FUEL STORAGE" (HI-STORM 100 FSAR TABLE 7.1.4), AS WELL AS THROUGH DEMONSTRATION OF A LEAKTIGHT CONDITION, THERE IS NO INCREASE IN FREQUENCY OF OCCURRENCE OF A CONFINEMENT BOUNDARY LEAKAGE.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the CFSAR?

   ☐ Yes
   ☒ No

BASIS:

The only malfunctions explicitly evaluated in the HI-STORM 100 FSAR are a malfunction of the forced helium dehydrator (FHD) and a power failure of the supplemental cooling system (SCS). Because the proposed activity has no impact on FHD or SCS operations, the likelihood of their malfunction is not affected by the proposed activity. The proposed activity strictly affects the port cover plates, and as demonstrated by Section I, all MPC design functions remain as described in the FSAR. Note that the port tubes and port caps are not required to be leaktight because they are not part of the confinement boundary, so any leakage past these components does not subject the MPC lid or port cover plate to additional loadings or increase the likelihood of their malfunction. No other SSCs important to safety are affected, and no FSAR-prescribed design limits are exceeded. Therefore, the proposed activity will not result in an increase in the likelihood of occurrence of a malfunction of an SSC previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the CFSAR? ☐ Yes ☒ No

BASIS:

Of the off-normal and accident conditions evaluated in the HI-STORM 100 FSAR, none are shown to cause unacceptable damage to the storage system, or the fuel contained therein. The proposed activity strictly affects the port cover plates, and as discussed in Section I, all MPC design functions remain as described in the FSAR. The initial helium backfill pressure remains compliant with the Technical Specifications, so the off-normal pressure condition remains as described in the FSAR. Confinement boundary leakage remains non-credible through continued compliance with DIVISION OF SPENT FUEL STORAGE AND TRANSPORTATION INTERIM STAFF GUIDANCE - ISG-18, REV. 1, “THE DESIGN AND TESTING OF LID WELDS ON AUSTENITIC STAINLESS STEEL CANISTERS AS THE CONFINEMENT BOUNDARY FOR SPENT FUEL STORAGE”, as well as demonstration of a leaktight condition in accordance with the HI-STORM 100 CoC and Technical Specifications. The shielding contribution of the MPC is maintained as a result of the proposed activity and no other SSCs important to safety are directly or indirectly affected, so the overall shielding performance of the system is maintained during any postulated accidents. Therefore, the proposed activity does not result in an increase in the consequences (i.e., controlled area boundary dose) of an accident previously evaluated in the CFSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the CFSAR? ☐ Yes ☒ No

BASIS:

THE ONLY MALFUNCTIONS EXPLICITLY EVALUATED IN THE HI-STORM 100 FSAR ARE A MALFUNCTION OF THE FHD AND A POWER FAILURE OF THE SCS. THE PROPOSED ACTIVITY STRICTLY AFFECTS THE PORT COVER PLATES, AND AS DISCUSSED IN SECTION I, ALL MPC DESIGN FUNCTIONS REMAIN AS DESCRIBED IN THE FSAR. THE SHIELDING CONTRIBUTION OF THE MPC IS MAINTAINED AS A RESULT OF THE PROPOSED ACTIVITY AND NO OTHER SSCS IMPORTANT TO SAFETY ARE DIRECTLY OR INDIRECTLY AFFECTED, SO THE OVERALL SHIELDING PERFORMANCE OF THE SYSTEM DUE TO AN FHD OR SCS MALFUNCTION IS MAINTAINED. THUS, THE PROPOSED ACTIVITY DOES NOT RESULT IN AN INCREASE IN THE CONSEQUENCES (I.E., CONTROLLED AREA BOUNDARY DOSE) OF A MALFUNCTION PREVIOUSLY EVALUATED IN THE FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the CFSAR? ☐ Yes ☒ No

BASIS:

The proposed activity strictly affects the port cover plates. As discussed in Section I, the proposed activity remains bounded by the structural evaluations performed in HI-2012787, while helium leak testing will confirm the confinement boundary complies with the leaktight criteria required by the HI-STORM 100 FSAR. Note that the port tubes and port caps are not required to be leaktight because they are not part of the confinement boundary, so any leakage past these components does not constitute a new accident. All MPC design functions remain as described in the FSAR. No other SSCs important to safety are affected and the proposed activity does not alter any operating procedures or handling methods. Therefore, the proposed activity will not create a possibility for an accident of a different type than any previously evaluated in the CFSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the CFSAR? ☐ Yes ☒ No

BASIS:

The proposed activity strictly affects the port cover plates (both vent or drain ports). As discussed in Section I, the port cover plates continue to meet the FSAR-prescribed stress limits, so the likelihood of their malfunction remains as described in the FSAR (i.e., non-credible). Note that the port tubes and port caps are not required to be leaktight because they are not part of the confinement boundary, so any leakage past these components does not introduce the possibility for a new malfunction. No other design functions of the MPC are affected and no other SSCs important to safety are affected, so the proposed activity does not cause any design limits in the FSAR to be exceeded. The proposed activity does not introduce any new materials to the HI-STORM 100 system, so there is no long-term material degradation threat to the system components. Therefore, the proposed activity will not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the CFSAR.

7. Result in a design basis limit for a fission product barrier as described in the CFSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

: The fission product barriers in the HI-STORM system are the cladding of the intact fuel assemblies and the MPC enclosure vessel confinement boundary. The critical design basis limit for the cladding of the intact fuel assemblies is the fuel cladding temperature. The critical design basis limits for the MPC enclosure vessel are the stresses and internal pressure.

The proposed activity strictly affects the port cover plates, and as discussed in Section I, the MPC confinement boundary continues to meet the FSAR-prescribed stress limits. The thermal performance of the MPC and the HI-STORM system are not affected, so MPC internal pressure and component temperatures (including fuel cladding) will not be increased beyond the FSAR design basis limits. As such, stresses in the MPC enclosure vessel will not be increased and the cladding of intact fuel assemblies will continue to hold contained gases without leakage. No change is being made to any design basis limit of the system. Therefore, no design basis limit for a fission product barrier is altered or exceeded.

8. Result in a departure from a method of evaluation described in the CFSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

THE PROPOSED ACTIVITY IS TO REPLACE THE PORT COVER PLATE (BOTH VENT AND DRAIN PORTS) IS NOT RELATED TO ANY DESIGN METHODOLOGY. ALL EXISTING EVALUATIONS DESCRIBED IN THE CFSAR REMAIN APPLICABLE AND NO NEW EVALUATIONS ARE NECESSARY. THEREFORE, THE PROPOSED ACTIVITY DOES NOT RESULT IN A DEPARTURE FROM A METHOD OF EVALUATION DESCRIBED IN THE CFSAR USED IN ESTABLISHING THE DESIGN BASES OR IN THE SAFETY ANALYSES.

**If any of the above questions is checked "Yes," request that the Certificate Holder process an amendment to the CoC and obtain NRC approval prior to implementing the change.**

**Attachment to**

**W3F1-2020-0032**

**Waterford 3 Summary of Commitment Changes**

| CCEF Number | Commitment Number | Commitment Description | Reason for Change/Deletion |
|---|---|---|---|
| CCEF-2018-005 | P-14897 | Failure to meet Technical Specification 3.3.3.11 Severity Level IV Violation. | CE-001-003 is being deleted. EN-CY-100 is the new implementing procedure. |
| CCEF-2019-005 | A-27654 | Submit a Licensee Amendment Request to the NRC by April 30, 2019 to request approval of a change to the existing TS 3.4.8.1 Figures 3.4-2 and 3.4-3 to incorporate the Capsule 83 test results as documented in report WCAP-17969-NP to allow operation past 32 EFPY. | Revise date from April 30, 2019 to June 30, 2023. |
| CCEF-2019-006 | A-27766 | For each nozzle with a full structural weld overlay installed, submit to the NRC required documents. | Changed list of required documents to be submitted to the NRC. |
| CCEF-2019-009 | A-27654 | Submit a License Amendment Request to the NRC by April 30, 2019 to request approval of a change to the existing TS 3.4.8.1 Figure 3.4-2 and Figure 3.4-3 to incorporate the capsule 83 test results as documented in report WCAP-17969-NP to allow operation past 32 EFPY. | Revise date of submittal to July 30, 2020. |