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

W3F1-2014-0028

April 29, 2014

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 25, 2014
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
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 26, 2012 through April 25, 2014 and includes copies of the 10CFR50.59 Evaluations from this period. However, this Submittal does not include a Summary report for 10CFR72.48 since there were none performed during 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 John Jarrell at (504) 739-6685.

There are no new commitments contained in this submittal.

Sincerely,

JPJ/LLB

Attachment: Summary of Evaluations
Enclosure: Waterford 3 10CFR 50.59 Evaluations
Commitment Change Summary Report

IE47
NRR

cc: Mr. Marc L. Depas, Regional Administrator — RidsRgn4MailCenter@nrc.gov
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Attachment

W3F1-2014-0028

Summary of Evaluations

10CFR50.59 Evaluation Number	Initiating Document	Summary
12-01	EC-0000030976 R000	SI-1239A(B) Backup Air Supply
12-02	CR-WF3-2012-3280	Documents Valve ACC-126A (CCW 'A' Temperature Control Valve) associated with procedure EP-002-100, Technical Support Center (TSC) Activation, Operation and Deactivation
10-09-01	EC-00000008439 R001	Miscellaneous Rupture Restraint Modification for SG/RVCH Replacement
12-03	ECN-0000040132 R000	(EC8458) Incorporation of RSG Design Document Updates and Review of Design Basis Methodologies for RSG Design Report
12-04	EC-0000030663 R000	W3C19 Core Reload 50.59 Evaluation
12-05	EC-0000041095 R000	Backup Air Supply System to Nitrogen Accumulators 1 and 2
12-06	EC-0000040444 R000	Impact of MSSV/ADV Leakage on Accident Dose Consequences
12-07	EC-0000025199 R000	GSI 191 (Generic Safety Issue) Margin Reduction for Safety Injection Sump
12-08	EC-0000041355 R000	Evaluate Manual Operator Actions Outside Control Room for Certain Air Operated Valves after Accumulator is Exhausted
13-01	EC-0000043821 R000	Safety Injection Tank 2B, Nitrogen Supply Temperature
10-06-01	EC-0000014765 R001	SI-405A9B) Bypass Fill/Equalization Line Addition/ECN-25944, changes for calculation MPR-2390 R3 SDC Gas Intrusion Analysis
14-01	EC-0000043927 R000	Vital and Instrument SUPS Upgrade Project

10CFR72.48 Evaluation Number	Initiating Document	Summary
None		

Enclosure to W3F1-2014-0028

Waterford 3 10CFR50.59 Evaluations and Commitment Change Summary Report

(172 pages)

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
2012-0001	P-27254	Commitment [P-27254]: Compensatory Measure 1; Safeguards Information, see attachment in W3F1-2009-0012, "Security Defensive Strategy" / March 22nd 2009 [Safeguards] Comments: LO-LAR-2009-00117, "CCEF" modified compensatory measure for number of compensatory posts, security positions.	Remove commitment; the commitment has been deemed unnecessary due to substantive changes in PSP compliance with revision to rule 73.55, actions implemented from RCA (CR-WF3-2009-1307) to include usage of CAP in drills and tabletops, and various successful drills and tabletops. PS-018-115, "Defensive Strategy and Tactical Deployment" Revision: 19 has been changed to incorporate a diverse reallocation of resources to ensure adequate protection and capabilities to detect, assess, interdict, and neutralize threats up to and including the design basis threat of radiological sabotage. NOTE: Detailed [Safeguards] justification can be found in document [SGI-WF3-2012-500] which is held by Security Management.
2012-0002	P-27228 & P-27076	In SFP Makeup Internal Strategy, W3 states that tie downs and hoses will be stored at +46 Fuel Handling Building. Revised Commitment Description: In SFP Makeup Internal Strategy, tie downs and hoses will be stored in the Grey B.5.b trailer.	Response not substantively impacted and change in line with S-SAMG-01 procedural guidance and training.
2012-0003	A-27017	Waterford 3 will continue to perform augmented inspections of the secondary side of the steam generators in each subsequent refueling outage until the current steam generators are replaced. The augmented inspections will include the upper batwing to wrapper bar welds, inspection of the stay cavity region,	During Refueling Outage 18, the Waterford 3 steam generators have been replaced. The action to inspect the batwings is no longer applicable.

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
		and foreign object search and retrieval (FOSAR) of the secondary side.	
2013-0001	P-21130	Waterford 3's Inservice testing plan revision 7 change 4 relief request 3.1.71 to allow the following Safety Injection check valves to be fully stroked each refueling outage and partially exercised quarterly. SI-241, SI-243, SI-510A, SI-510B, SI-512A, SI-512B, SI-242, and SI-244.	Waterford 3 Procedure SEP-WF3-IST-2 documents the IST program. For the Waterford 3 IST Plan, the applicable Code is the ASME OM Code-2001 Edition with addenda through and including the ASME OMb Code-2003 Addenda (referred to as OMb-2003). This code has superseded the code in effect when the reference relief request was issued. Relief request 3.1.71 is no longer in effect. Valve open and closed stroke tests are based on the population of valves in each group and the valve performance history. The frequency of valve testing is controlled by the IST program. This commitment has been superseded by a change to the IST code and can be cancelled.
2013-0002	P21140	Waterford 3's Inservice testing plan revision 7 change 4 relief request 3.1.14 to allow the following Safety Injection check valves to be fully stroked each refueling outage and partially exercised quarterly. SI-207A, SI-207B, SI-207AB & SI-216	Waterford 3 Procedure SEP-WF3-IST-2 documents the IST program. For the Waterford 3 IST Plan, the applicable Code is the ASME OM Code-2001 Edition with addenda through and including the ASME OMb Code-2003 Addenda (referred to as OMb-2003). This code has superseded the code in effect when the reference relief request was issued. Relief request 3.1.71 is no longer in effect. Valve open and closed stroke tests are based on the population of valves in each group and the valve performance history. The frequency of valve testing is controlled by the IST program. This commitment has been superseded by a change to the IST code and can be cancelled.

I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # / Rev. #:** 2012-01 / Rev. 0

Proposed Change / Document: EC-30976, SI-129A(B) Backup Air Supply

Description of Change:

Engineering Change (EC)-30976 adds an alternate method of controlling flow rate through the Shutdown Cooling (SDC) Heat Exchanger (HX) and the cooldown rate of the Reactor Coolant System (RCS) during SDC in the postulated event of an extended loss of Instrument Air (IA) where the safeguard pump rooms are not accessible because of radiation dose rates. The screening justifies that the change does not adversely affect the design function, method of performing or controlling the design function, or method of evaluation that demonstrates the design function will be accomplished. However, since the proposed change introduces a new manual method of accomplishing design function of performing a cooldown of RCS by forcing closed the shutdown cooling flow control valve with manually aligned alternate air and controlling shutdown cooling flow with the shutdown cooling heat exchanger temperature control valve (SI-415), the change is being evaluated under the 50.59 rule.

Specific Changes

EC-30976 adds a backup air supply capable of closing SDC Flow Control Valves, SI-129A(B), LPSI A(B) Flow Control Valve, from the -15 Reactor Auxiliary Building (RAB) Valve Gallery without entering the safeguard pump room. The backup air supply system will port air from a new storage accumulator into the top cylinder of the SI-129A(B) actuator while simultaneously venting air from the bottom cylinder of the actuator, causing the SI-129A(B) valve to close and all SDC flow to pass through the SDC Heat Exchanger. Reference drawing G167 sheet 3 for illustration. The air supply for each valve will consist of one air accumulator, pressure indicator, an isolation valve, tubing, a vent valve, and pneumatically actuated shuttle valves. Reference drawing B430 sheet V03a for illustration. The backup air supply accumulator will normally be isolated from the tubing

[1] 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.

and actuator. Accumulator pressure will be monitored as part of Operations Shift Logs, OI-004-000. Maximum accumulator pressure will be less than the design pressure of the SI-129A(B) actuator components. Periodic leak testing and functional testing will be performed on the tubing, shuttle valves, and the SI-129A(B) actuator with STA-001-005, Leakage Testing of Air and Nitrogen Accumulators for Safety Related Valves. After SI-129A(B) closes, no additional air is required to maintain it in the closed position, based on actuator torque calculation, ECM97-069. A vent valve is provided to reduce pressure in the new backup air supply piping to restore normal functionality when the Instrument Air (IA) system is restored. The backup air system tubing will be normally vented to eliminate any chance of a leaking valve causing inadvertent closure of SI-129A(B).

In the event that the safeguard pump rooms are not accessible due to high radiation levels, rather than locally manually throttling SI-129A(B) using the handwheel to adjust shutdown cooling flow, SI-129A(B) will be locally closed using the new backup air supply from the -15 RAB Valve Gallery. Total SDC flow will be controlled remotely with the SDC Heat Exchanger Temperature Control Valves, SI-415A(B). The analysis demonstrates that, with Shutdown Cooling HX A(B) CCW Flow Control Valve, CC-963A(B) fully open, CCW piping temperature downstream of the SDC HX is less than the rated temperature of 270°F for RCS temperatures less than 350°F and that RCS cooldown limit of 100°F/hr will not be exceeded. Therefore, it is acceptable to operate SDC with SI-129A(B) closed, SI-415A(B) throttled to achieve 4000 gpm SDC flow, and CC-963A(B) fully open for the duration of the event after SDC entry conditions are achieved. The analysis assumes only one train of SDC is initially placed in service. Two trains may be placed in service without throttling of CC-963A(B) and without exceeding allowable cooldown rate of 100°F/hr when RCS temperature is ≤230°F. Post Return to Service Actions are established to track procedure changes and training necessary to protect the assumptions in the analysis.

The existing local manual control functionality (handwheel) is not affected and may be available depending on the dose rates in the safeguard pump rooms where SI-129A(B) is located. The existing remote control functionality is not affected when Instrument Air remains available.

Background

The design function of the SDC system is to remove decay heat at lower RCS temperature and pressure. FSAR Section 9.3.6.3.3 states that the system is designed to reduce RCS temperature to cold shutdown temperature of 200°F in approximately 20 hours. Normally, when SDC is commenced, air operated valve SI-129A(B), SDC Flow Control Valve, is switched in the control room from OPEN to AUTO and modulates to maintain the selected total SDC flow rate while motor operated valve SI-415A(B), SDC Heat Exchanger Temperature Control Valves, is adjusted to control cool down rate by throttling flow through the SDC Heat Exchanger. Air operated valve CC-963A(B) is placed in the OPEN position to ensure at least 2305 gpm Component Cooling Water (CCW) flow through the SDC Heat Exchanger to protect the design temperature basis of the CCW piping. CC-963A(B) fails open on loss of IA.

SDC Flow Control Valve, SI-129A(B), which controls bypass flow around the SDC Heat Exchanger, fails open on loss of IA. SI-129A(B) is normally open in the safety injection lineup and has a safety related function to remain open for Low Pressure Safety Injection (LPSI) flow. SI-129A(B) does not have a safety related air supply capable of throttling SDC Heat Exchanger bypass flow rate. If SI-129A(B) fails open, then Reactor Coolant (RCS) flow through the SDC Heat Exchanger is only 1,012 gpm per "SIS Split Flow Calculation", MNQ6-48. This flow rate is not adequate to remove decay heat and obtain/maintain 200°F RCS temperature until about 272 hours after the reactor is subcritical per "Decay Heat Removal with Reduced Flow Through the Shutdown Cooling Heat Exchanger", ECM11-003. General Design Criterion 34, Residual Heat Removal, requires the system that removes residual heat to be designed to safety grade requirements. The original design compensated for SI-129A(B) failing open by steaming through Atmospheric Dump Valves (ADV) to remove residual heat for 21 hours after an event and then proceeding with a slow cooldown. Before Extended Power Uprate (EPU), decay heat was low enough for SDC to maintain the RCS at 350°F with SI-129A(B) failed open, after about 21 hours. CR-WF3-2009-3030 identifies that ECS04-013, Small Break Loss of Coolant Accident (SBLOCA) Alternate Source Term (AST) Radiological Dose Consequences Analysis, was revised with Extended Power Uprate (EPU) to credit going on SDC by 7.5 hours after an event. Delaying the initiation of SDC will have an adverse affect on the current analyses for control room and offsite dose consequences and water inventories.

In the event of a loss of IA, the current license basis credits local handwheel operation of SI-129A(B). Currently, FSAR section 9.3.6.3.1 states: "A loss of instrument air to the SDCS will not result in a loss of cooling ability. The air operated shutdown cooling heat exchanger bypass valves are equipped with a hand wheel, which permits positioning upon loss of instrument air." CR-WF3-2009-3030 also identified that the AST SBLOCA analysis performed for EPU predicts dose rates in the safeguard pump rooms that would make them inaccessible.

Currently FSAR section 9.3.6.3.2 states: "For post-LOCA SDCS operation, a loss of air is assumed for flow control valves SI-306 (SI-129B) and SI-307 (SI-129A) since the air supplies for these valves are not seismically qualified. Under these circumstances, valves SI-306 (SI-129B) and SI-307 (SI-129A) fail open and cannot be remotely controlled from the main control room. The shutdown cooling process is then controlled by adjusting the flow rate of reactor coolant through the shutdown cooling heat exchangers remotely from the main control room with throttle valves SI-656 (SI-415B) and SI-657 (SI-415A) and adjusting total shutdown cooling flow locally by manually positioning valves SI-306 (SI-129B) and SI-307 (SI-129A)."

FSAR Section 9.3.6.3.3 describes the SDC system performance evaluation. The license basis credits that the RCS can be brought to 200°F in approximately 20 hours. With SI-129A(B) failed open, RCS temperature of 200°F cannot be maintained until much longer, approximately 272 hours.

As described in FSAR Section 9.3.6.2.1, "during initial cooldown, the temperature differences for heat transfer are large, thus only a portion of the total shutdown flow is diverted through the heat exchangers. As cooldown proceeds, the temperature differences become less and the flow rate through the heat exchangers is increased."

Reasoning:

This 50.59 Evaluation is performed by answering the eight questions in section II with the following additional considerations addressing the potentially adverse impact of the change to the operator action required to throttle SI-129A(B) during the unlikely event of requiring shutdown cooling operation coincident with a loss of instrument air where the safeguard pump rooms are not accessible.

In the event of a loss of instrument air, where the safeguard pump rooms are not

accessible due to high radiation levels, rather than locally manually throttling SI-129A(B) using the handwheel to adjust shutdown cooling flow, SI-129A(B) will be locally closed using the new backup air supply from the -15 RAB Valve Gallery. Total SDC flow will be controlled remotely with the SDC Heat Exchanger Temperature Control Valves, SI-415A(B). The analysis demonstrates that, with Shutdown Cooling HX A(B) CCW Flow Control Valve, CC-963A(B) fully open, CCW piping temperature downstream of the SDC HX is less than the rated temperature of 270°F for RCS temperatures less than 350°F and that RCS cooldown limit of 100°F/hr will not be exceeded. Therefore, it is acceptable to operate SDC with SI-129A(B) closed, SI-415A(B) throttled to achieve 4000 gpm SDC flow, and CC-963A(B) fully open for the duration of the event after SDC entry conditions are achieved. The analysis assumes only one train of SDC is initially placed in service. Two trains may be placed in service without throttling of CC-963A(B) and without exceeding allowable cooldown rate of 100°F/hr when RCS temperature is ≤230°F. Post Return to Service Actions are established to track procedure changes and training necessary to protect the assumptions in the analysis.

The existing local manual control functionality (handwheel) is not affected and may be available depending on the dose rates in the safeguard pump rooms where SI-129A(B) is located. The existing remote control functionality is not affected when Instrument Air remains available.

Technical Specification 3.4.1.5 requires two shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation. The APPLICABILITY is MODE 5 with reactor coolant loops not filled. The TS definition of OPERABLE/OPERABILITY is a system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

If an operator action must be performed prior to a system being capable of performing its specified safety function, then it must be evaluated with respect to the guidance presented in NRC Information Notice 97-78, Crediting of Operator Actions In Place of Automatic Actions and Modifications of Operator Actions, Including Response Times, October 23, 1997, NRC Regulatory Issue Summary 2005-20, Revision To NRC Inspection Manual Part 9900

Technical Guidance, Operability Determinations & Functionality Assessments For Resolution Of Degraded Or Nonconforming Conditions Adverse To Quality Or Safety, April 16, 2008, and NRC Inspection Manual Technical Guidance Part 9900 ITSB, Operability Determinations & Functionality Assessments For Resolution Of Degraded Or Nonconforming Conditions Adverse To Quality Or Safety.

NRC INFORMATION NOTICE 97-78 lists the following requirements for crediting an operator action:

> The original design of nuclear power plant safety systems and their ability to respond to design-basis accidents were described in licensees' FSARs and were reviewed and approved by the NRC. Most safety systems were designed to rely on automatic system actuation to ensure that the safety systems were capable of carrying out their intended functions. In a few cases, limited operator actions, when appropriately justified, were approved. Proposed changes that substitute manual action for automatic system actuation or modify existing operator actions, including operator response times, previously reviewed and approved during the original licensing review of the plant will, in all likelihood, raise the possibility of a USQ. Such changes must be evaluated under the criteria of 10 CFR 50.59 to determine whether a USQ is involved and whether NRC review and approval is required before implementation. A licensee may not make such changes before it receives approval from the NRC when the change, test, or experiment may (1) increase the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously analyzed in the FSAR, (2) create the possibility of an accident or a malfunction of a different type than any previously evaluated in the FSAR, or (3) reduce the margin of safety as defined in the basis for any TS. In the NRC staff's experience, many of the changes of the type described above proposed by licensees do involve a USQ.

NRC INFORMATION NOTICE 97-78 also lists specific requirements the NRC will use to review new operator actions. Based on these guidelines, the NRC's reviews of licensees' analyses typically include, but are not limited to, (1) the specific operator actions required; (2) the potentially harsh or inhospitable environmental conditions expected; (3) a general discussion of the ingress/egress paths taken by the operators to accomplish functions; (4) the

procedural guidance for required actions; (5) the specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions; (6) any additional support personnel and/or equipment required by the operator to carry out actions; (7) a description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken; (8) the ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery; and (9) consideration of the risk significance of the proposed operator actions.

NRC REGULATORY ISSUE SUMMARY 2005-20 issued a new version of NRC Inspection Manual Technical Guidance Part 9900 ITSB. The NRC Inspection Manual lists the following requirements for crediting an operator action:

> For situations where substitution of manual action for automatic action is proposed for an operability determination, the evaluation of manual action must focus on the physical differences between automatic and manual action and the ability of the manual action to accomplish the specified safety function or functions. The physical differences to be considered include the ability to recognize input signals for action, ready access to or recognition of setpoints, design nuances that may complicate subsequent manual operation (such as auto-reset, repositioning on temperature or pressure), timing required for automatic action, minimum staffing requirements, and emergency operating procedures written for the automatic mode of operation. The licensee should have written procedures in place and personnel should be trained on the procedures before any manual action is substituted for the loss of an automatic action.
>
> The assignment of a dedicated operator for a manual action requires written procedures and full consideration of all pertinent differences. The consideration of a manual action in remote areas must include the abilities of the assigned personnel and how much time is needed to reach the area, training of personnel to accomplish the task, and occupational hazards such as radiation, temperature, chemical, sound, or visibility hazards. One reasonable test of the reliability and effectiveness of a manual action may be the approval of the manual action for the same function at a similar facility.

The manual operator action was evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994, American National Standard Time Response Design Criteria for Safety Related Operator Actions, August 23, 1994, and is addressed in Question #5.

Summary of Evaluation:

This evaluation determined that using a backup air supply to throttle closed SI-129A(B) rather than using the local handwheel does not represent any unreviewed safety question and does not require prior NRC review and approval.

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: Dale Gallodoro / See EC 30976 / EOI / Design Engineering / 04-17-12
Name (print) / Signature / Company / Department / Date

Reviewer: James Tinkle / See EC 30976 / Iepson / Design Engineering / 04-19-12
Name (print) / Signature / Company / Department / Date

OSRC: See EC-30976
Chairman's Name (print) / Signature / Date

William McKinney / See EC 30976 / 04/25/2012
OSRC Meeting # W3 12-04

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: THIS CHANGE AFFECTS THE SHUTDOWN COOLING FLOW CONTROL VALVE, SI-129A(B) AND A NEW SAFETY RELATED BACKUP INSTRUMENT AIR SYSTEM DESIGNED TO FORCE SI-129A(B) CLOSED. THE CHAPTER 15 ACCIDENTS EVALUATED IN THE UFSAR THAT MAY BE AFFECTED BY THE OPERATION OF SI-129A(B) ARE LOSS OF COOLANT ACCIDENT (LOCA) (FSAR 15.6.3.3), STEAM GENERATOR TUBE RUPTURE (SGTR) (FSAR 15.6.3.2), AND OTHER EVENTS THAT EVENTUALLY ACHIEVE SDC ENTRY CONDITIONS. THE INSTRUMENT AIR SYSTEM AND SHUTDOWN COOLING SYSTEM ARE NOT INITIATORS OF ANY ACCIDENT PREVIOUSLY EVALUATED IN THE FSAR. SI-129A(B) IS NORMALLY OPEN TO PROVIDE A FLOWPATH FOR LPSI IN THE EVENT OF A SAFETY INJECTION ACTUATION. A BACKUP AIR SUPPLY CAPABLE OF CLOSING SI-129A(B) VALVE FROM THE -15 VALVE GALLERY FOR SDC OPERATION IS BEING INSTALLED. THE SYSTEM WILL BE CONTROLLED BY A 3-WAY BALL VALVE, WHICH WILL NORMALLY BE LOCKED TO ISOLATE THE SYSTEM, PREVENTING INADVERTENT CLOSURE OF THE SI-129A(B) VALVE. OPENING OR CLOSING THE SI-129A(B) VALVE IS NOT AN ACCIDENT INITIATOR. THE CHANGE OF PLANT OPERATOR ACTION FROM MANUALLY OPENING THE SI-129A(B) VALVE DURING LOSS OF IA COINCIDENT WITH THE NEED FOR SHUTDOWN COOLING TO: OPENING THE ACCUMULATOR VALVE TO REMOTELY OPEN THE SI-129A(B) VALVE TO PROVIDE SHUTDOWN COOLING DOES NOT RESULT IN MORE THAN A MINIMAL INCREASE IN THE FREQUENCY OF AN OCCURRENCE OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR. THE OPERATOR ACTION WILL BE IN RESPONSE TO LOSS OF INSTRUMENT AIR THAT HAS AN EXISTING PLANT ANALYSIS RESPONSE THAT REMAINS UNCHANGED. 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: EC-30976 ADDS AN ALTERNATE METHOD OF CONTROLLING SHUTDOWN COOLING (SDC) FLOW RATE AND COOLDOWN RATE IN THE POSTULATED EVENT OF AN EXTENDED LOSS OF INSTRUMENT AIR WHERE THE SAFEGUARD PUMP ROOMS ARE NOT ACCESSIBLE BECAUSE OF RADIATION DOSE RATES. EC-30976 ADDS A BACKUP AIR SUPPLY

CAPABLE OF CLOSING SDC FLOW CONTROL VALVES, SI-129A(B), FROM THE -15 RAB VALVE GALLERY WITHOUT ENTERING THE SAFEGUARD PUMP ROOM.

THE BACKUP AIR SUPPLY SYSTEM WILL, UPON MANUAL LOCAL OPERATOR ACTION, ALIGN AIR FROM A BACKUP AIR SUPPLY ACCUMULATOR INTO THE TOP CYLINDER OF THE SI-129A(B) ACTUATOR WHILE SIMULTANEOUSLY VENTING AIR FROM THE BOTTOM CYLINDER OF THE ACTUATOR. THE AIR SUPPLY FOR EACH VALVE WILL CONSIST OF ONE AIR ACCUMULATOR, A THREE WAY BALL VALVE, TUBING, A VENT VALVE, MAINTENANCE ISOLATION VALVES, AND PNEUMATICALLY ACTUATED SHUTTLE VALVES. THE BACKUP AIR SUPPLY ACCUMULATOR WILL NORMALLY BE ISOLATED FROM THE TUBING AND SI-129A(B) ACTUATOR WITH A LOCKED 3-WAY BALL VALVE. ACCUMULATOR PRESSURE WILL BE MONITORED AS PART OF OPERATOR ROUNDS. MAXIMUM BOTTLE PRESSURE WILL BE LESS THAN THE DESIGN PRESSURE OF THE SI-129A(B) ACTUATOR COMPONENTS. PERIODIC LEAK TESTING AND FUNCTIONAL TESTING WILL BE PERFORMED ON THE TUBING, SHUTTLE VALVES, AND THE SI-129A(B) ACTUATOR. THE BACKUP AIR SUPPLY SYSTEM IS DESIGNED TO THE RULES OF ANSI B31.1 WITH SEISMIC I QUALIFICATION. TUBING AND BALL VALVES WILL BE SUPPLIED IN ACCORDANCE WITH ASME SECTION III. IN ACCORDANCE WITH B430 GENERAL NOTES 57A AND 57B, ACCESSORY COMPONENTS, FOR WHICH THE ASME CODE WAS NOT INTENDED TO BE APPLIED MAY MEET THE ORIGINAL DESIGN REQUIREMENTS, INCLUDING ANSI B31.1, SEISMIC QUALIFICATION, AND 10CFR50 APPENDIX B COMPLIANCE. THEREFORE, THE NEW BACKUP AIR SUPPLY WILL MEET ORIGINAL DESIGN SPECIFICATIONS APPLICABLE TO THE SI-129A(B) VALVE OPERATOR AND OTHER SAFETY RELATED VALVE TOPWORKS. THE DESIGN MAINTAINS SEPARATION AND REDUNDANCY BY USING A SEPARATE BACKUP AIR SUPPLY SYSTEM FOR EACH TRAIN.

AFTER SI-129A(B) CLOSES, NO ADDITIONAL AIR IS REQUIRED TO MAINTAIN IT IN THE CLOSED POSITION BASED ON ACTUATOR TORQUE CALCULATION ECM97-069, WHICH SHOWS ACTUATOR TORQUE IS REQUIRED TO OPEN SI-129A(B) UNDER DIFFERENTIAL PRESSURE CONDITIONS. THE 3-WAY VALVE WLL VENT PRESSURE IN THE NEW BACKUP AIR SUPPLY PIPING TO RESTORE NORMAL FUNCTIONALITY WHEN THE INSTRUMENT AIR (IA) SYSTEM IS RESTORED. THE CURRENT DESIGN CREDITS THROTTLING SI-129A(B) WITH THE LOCAL HANDWHEEL. CURRENTLY, FSAR SECTION 9.3.6.3.1 STATES: "A LOSS OF INSTRUMENT AIR TO THE SDCS WILL NOT RESULT IN A LOSS OF COOLING ABILITY. THE AIR OPERATED SHUTDOWN COOLING HEAT EXCHANGER BYPASS VALVES ARE EQUIPPED WITH A HAND WHEEL, WHICH PERMITS POSITIONING UPON LOSS OF INSTRUMENT AIR." HOWEVER, CR-WF3-2009-3030 IDENTIFIES THAT THE VALVE MAY NOT BE ACCESSIBLE IN THE EVENT OF A SBLOCA. RATHER THAN LOCALLY MANUALLY THROTTLING SI-129A(B) USING THE HANDWHEEL TO ADJUST SHUTDOWN COOLING FLOW, THIS EC WILL ALLOW SI-129A(B) TO BE LOCALLY CLOSED USING THE NEW BACKUP AIR SUPPLY FROM THE -15 RAB VALVE GALLERY, WHICH ALREADY NEEDS TO BE ACCESSED TO CLOSE THE CONTAINMENT SPRAY, CS-117A(B) VALVES TO ALIGN SDC. THE ALIGNMENT INVOLVES UNLOCKING ONE VALVE AND TURNING ITS HANDLE 90 DEGREES, WHICH IS FULLY WITHIN CAPABILITIES OF TRAINED OPERATIONS PERSONNEL USING PROCEDURAL GUIDANCE AND IS EXPECTED TO REQUIRE LESS IMPLEMENTING TIME THAN MANUALLY OPERATING SI-129 IN THE SAFEGUARDS PUMP

ROOM. TOTAL SDC FLOW WILL BE CONTROLLED REMOTELY WITH THE SDC HEAT EXCHANGER TEMPERATURE CONTROL VALVES, SI-415A(B). THE ANALYSIS IN ECM11-003 DEMONSTRATES THAT, WITH CC-963A(B) FULLY OPEN (I.E. DUE TO LOSS OF INSTRUMENT AIR), CCW PIPING TEMPERATURE DOWNSTREAM OF THE SDC HX IS LESS THAN THE RATED TEMPERATURE OF 270°F FOR RCS TEMPERATURES LESS THAN 350°F AND THAT RCS COOLDOWN LIMIT OF 100°F/HR WILL NOT BE EXCEEDED. THEREFORE, IT IS ACCEPTABLE TO OPERATE SDC WITH SI-129A(B) CLOSED, SI-415A(B) THROTTLED TO ACHIEVE 4000 GPM SDC FLOW, AND CC-963A(B) FULLY OPEN FOR THE DURATION OF THE EVENT AFTER SDC ENTRY CONDITIONS ARE ACHIEVED. THE ANALYSIS ASSUMES ONLY ONE TRAIN OF SDC IS INITIALLY PLACED IN SERVICE. TWO TRAINS MAY BE PLACED IN SERVICE WITHOUT THROTTLING OF CC-963A(B) AND WITHOUT EXCEEDING ALLOWABLE COOLDOWN RATE OF 100°F/HR WITH RCS TEMPERATURE IS ≤230°F. FSAR 9.3.6.2.1 CURRENTLY CREDITS THE ABILITY TO COOLDOWN WITH A SINGLE TRAIN OF SDC. EC-30976 INITIATES ACTIONS TO UPDATE OPERATING PROCEDURES AND PROVIDE TRAINING TO OPERATIONS PERSONNEL IMPLEMENTING SDC TO REFLECT THE CHANGES DESCRIBED ABOVE.

THE EXISTING LOCAL MANUAL CONTROL FUNCTIONALITY IS NOT AFFECTED AND MAY BE AVAILABLE DEPENDING ON THE DOSE RATES IN THE SAFEGUARD PUMP ROOMS WHERE SI-129A(B) IS LOCATED. THE EXISTING REMOTE CONTROL FUNCTIONALITY IS NOT AFFECTED WHEN INSTRUMENT AIR REMAINS AVAILABLE.

CURRENTLY, FSAR SECTION 9.3.6.3.1 STATES: "A LOSS OF INSTRUMENT AIR TO THE SDCS WILL NOT RESULT IN A LOSS OF COOLING ABILITY. THE AIR OPERATED SHUTDOWN COOLING HEAT EXCHANGER BYPASS VALVES ARE EQUIPPED WITH A HAND WHEEL. WHICH PERMITS POSITIONING UPON LOSS OF INSTRUMENT AIR."

THE NEW METHOD OF MANUALLY LOCALLY CONTROLLING SI-129A(B), SDC HEAT EXCHANGER FLOW RATE, AND RCS COOLDOWN RATE IS THE SAME AS THAT USING THE METHOD IN THE EXISTING LICENSE BASIS. BOTH REQUIRE LOCAL MANUAL CONTROL. THE NEW ALTERNATE METHOD IS LESS DIFFICULT, REQUIRES LESS TRAVEL, IS ACCOMPLISHED IN A LESS HARSH ENVIRONMENT, AND CAN BE ACCOMPLISHED IN LESS TIME, WITH THE SAME NUMBER OF OPERATORS.

THE CHAPTER 15 ACCIDENTS EVALUATED IN THE UFSAR THAT MAY BE AFFECTED BY A MALFUNCTION OF SI-129A(B) ARE LOCA (FSAR 15.6.3.3), STEAM GENERATOR TUBE RUPTURE (SGTR) (FSAR 15.6.3.2), AND OTHER EVENTS THAT EVENTUALLY ACHIEVE SDC ENTRY CONDITIONS. THE POTENTIAL MALFUNCTIONS INTRODUCED BY MODIFYING SI-129A(B) ARE 1) INADVERTENT CLOSURE, WHICH WOULD PREVENT LPSI FLOW, AND 2) FAILURE TO THROTTLE FOR SDC, WHICH WOULD RESULT IN HAVING TO STEAM LONGER TO REMOVE DECAY HEAT.

FSAR TABLE 6.3-1 LISTS THE FAILURE MODES AND EFFECTS ANALYSIS (FMEA) FOR SI-129A(B) (2SI-FM317A / 2SI-FM348B). THE FMEA POSTULATES AN OPERATOR ERROR THAT CAUSES THE VALVE TO BE CLOSED RESULTING IN A LOSS OF ONE LPSI TRAIN. THE INHERENT COMPENSATING PROVISION REMAINS THE REDUNDANT PARALLEL LPSI TRAIN WITH THE PROPOSED CHANGE. THIS POTENTIAL OUTCOME IS MINIMIZED BY OPERATIONS

TRAINING, OPERATIONS PROCEDURAL CONTROL FOR MAINTAINING POSITION OF BACKUP AIR SUPPLY VALVES AND HAVING ISOLATION CONTROLLED VIA A LOCK TO PRECLUDE INADVERTENT OPERATION.

THE ACTION OF ALIGNING THE BACKUP AIR SYSTEM TO SI-129 WILL BE REFLECTED IN OPERATIONS PROCEDURES AND OPERATOR TRAINING PROGRAMS. WATERFORD HAS EVALUATED IN EC-30976 THAT THIS MANUAL ACTION CAN BE ACCOMPLISHED IN LESS TIME THAN LOCAL MANUAL OPERATION VIA HANDWHEEL AND AREA OF -15 VALVE GALLERY IS FULLY ACCESSIBLE IN ACCIDENT SCENARIOS OF APPLICABLE SAFETY ANALYSIS. WATERFORD HAS REVIEWED THE ABILITY TO RECOVER FROM CREDIBLE ERRORS IN PERFORMANCE OF MANUAL ACTION IN THE ALIGNMENT OF ALTERNATE AIR AND CREDIBLE ERROR IS UNLIKELY SINCE ALIGNMENT OF ALTERNATE AIR SIMPLY CONSISTS OF UNLOCKING 3 WAY ISOLATION VALVE AND ALIGNING TO PLACE AIR RECEIVER IN SERVICE TO SI-129; FOR UNLIKELY EVENT OF HAVING ALTERNATE AIR ALIGNED WHEN SAFETY INJECTION REQUIRED, THERE ARE TWO TRAINS OF SAFETY INJECTION AND INDICATION IN CONTROL ROOM FOR SI-129 BEING OPEN. DESIGN OF SI-129 ALTERNATE AIR IS CONFIGURED SUCH THAT ONLY SI-129 IS IMPACTED BY MODIFICATION AND OPERATION OF ALIGNMENT; NO OTHER SSC IS IMPACTED .

THEREFORE, ENGINEERED AND ADMINISTRATIVE CONTROLS ARE SUFFICIENT TO ENSURE THAT POTENTIAL MALFUNCTIONS ARE NOT INTRODUCED. THEREFORE, THE PROPOSED CHANGE DOES NOT 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.

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

BASIS: EC-30976 ADDS AN ALTERNATE METHOD OF CONTROLLING SHUTDOWN COOLING (SDC) FLOW RATE AND COOLDOWN RATE IN THE POSTULATED EVENT OF AN EXTENDED LOSS OF INSTRUMENT AIR WHERE THE SAFEGUARD PUMP ROOMS ARE NOT ACCESSIBLE BECAUSE OF RADIATION DOSE RATES. EC-30976 ADDS A BACKUP AIR SUPPLY CAPABLE OF CLOSING SDC FLOW CONTROL VALVES, SI-129A(B), FROM THE -15 RAB VALVE GALLERY WITHOUT ENTERING THE SAFEGUARD PUMP ROOM. THE CHAPTER 15 ACCIDENTS EVALUATED IN THE UFSAR THAT MAY BE AFFECTED BY A MALFUNCTION OF SI-129A(B) ARE LOCA (FSAR 15.6.3.3), STEAM GENERATOR TUBE RUPTURE (SGTR), AND OTHER EVENTS THAT EVENTUALLY ACHIEVE SDC ENTRY CONDITIONS. ECS04-013, SBLOCA AST RADIOLOGICAL DOSE CONSEQUENCES ANALYSIS, WAS REVISED WITH EPU TO CREDIT GOING ON SDC BY 7.5 HOURS AFTER AN EVENT. DELAYING THE INITIATION OF SDC WILL HAVE AN ADVERSE AFFECT ON THE CURRENT ANALYSES FOR CONTROL ROOM AND OFFSITE DOSE CONSEQUENCES AND WATER INVENTORIES. THE POTENTIAL MALFUNCTIONS INTRODUCED BY MODIFYING SI-129A(B) ARE 1) INADVERTENT CLOSURE, WHICH WOULD PREVENT LPSI FLOW, AND 2) FAILURE TO THROTTLE FOR SDC, WHICH WOULD RESULT IN HAVING TO STEAM LONGER TO REMOVE DECAY HEAT. EC-30976 ADDS AN ALTERNATE METHOD OF CONTROLLING SHUTDOWN COOLING (SDC) FLOW RATE AND COOLDOWN RATE IN THE POSTULATED EVENT OF AN EXTENDED LOSS OF INSTRUMENT AIR WHERE

THE SAFEGUARD PUMP ROOMS ARE NOT ACCESSIBLE BECAUSE OF RADIATION DOSE RATES. THE BACKUP AIR SUPPLY ENABLES SDC INITIATION (SECURE STEAMING) WITHIN THE TIMEFRAME ASSUMED IN THE DOSE ANALYSES IN THE UNLIKELY EVENT OF A LOSS OF INSTRUMENT AIR WHILE THE SAFEGUARD PUMP ROOMS ARE INACCESSIBLE. ENGINEERED AND ADMINISTRATIVE CONTROLS ARE DESIGNED TO ENSURE THAT POTENTIAL MALFUNCTIONS ARE NOT INTRODUCED. THEREFORE, THE PROPOSED CHANGE DOES NOT RESULT IN MORE THAN A MINIMAL INCREASE IN THE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR.

THE ALTERNATE AIR MODIFICATION TO SI-129 HAS NO IMMEDIATE IMPACT TO ANY FISSION PRODUCT BARRIERS AND IS SOLELY PART OF A MITIGATIVE SYSTEM FUNCTION (CORE COOLING) THAT IS BEING ENHANCED TO FURTHER MINIMIZE DOSE TO CONTROL ROOM OPERATORS IN SCENARIO OF LOSS OF INSTRUMENT AIR DURING ACCIDENT RECOVERY.

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: EC-30976 ADDS AN ALTERNATE METHOD OF CONTROLLING SHUTDOWN COOLING (SDC) FLOW RATE AND COOLDOWN RATE IN THE POSTULATED EVENT OF AN EXTENDED LOSS OF INSTRUMENT AIR WHERE THE SAFEGUARD PUMP ROOMS ARE NOT ACCESSIBLE BECAUSE OF RADIATION DOSE RATES. THIS CHANGE AFFECTS THE SHUTDOWN COOLING FLOW CONTROL VALVE, SI-129A(B) AND A NEW SAFETY RELATED BACKUP INSTRUMENT AIR SYSTEM. SI-129A(B) IS NORMALLY OPEN TO PROVIDE A FLOWPATH FOR LPSI. A BACKUP AIR SUPPLY CAPABLE OF CLOSING SI-129A(B) VALVE FROM THE -15 VALVE GALLERY FOR SDC OPERATION IS BEING INSTALLED. THE SYSTEM WILL BE CONTROLLED BY A 3-WAY BALL VALVE, WHICH WILL NORMALLY BE LOCKED TO ISOLATE THE SYSTEM, PREVENTING INADVERTENT CLOSURE OF THE SI-129A(B) VALVE. OPENING OR CLOSING THE SI-129A(B) VALVE IS NOT AN ACCIDENT INITIATOR. THE CHAPTER 15 ACCIDENTS EVALUATED IN THE UFSAR THAT MAY BE AFFECTED BY A MALFUNCTION OF SI-129A(B) ARE LOCA (FSAR 15.6.3.3), STEAM GENERATOR TUBE RUPTURE (SGTR) (FSAR 15.6.3.2), AND OTHER EVENTS THAT EVENTUALLY ACHIEVE SDC ENTRY CONDITIONS. THE POTENTIAL MALFUNCTIONS INTRODUCED BY MODIFYING SI-129A(B) ARE 1) INADVERTENT CLOSURE, WHICH WOULD PREVENT LPSI FLOW, AND 2) FAILURE TO THROTTLE FOR SDC, WHICH WOULD RESULT IN HAVING TO STEAM LONGER TO REMOVE DECAY HEAT. THE BACKUP AIR SUPPLY ENABLES SDC INITIATION WITHIN THE TIMEFRAME ASSUMED IN THE DOSE ANALYSES. ENGINEERED AND ADMINISTRATIVE CONTROLS ARE DESIGNED TO ENSURE THAT POTENTIAL MALFUNCTIONS ARE NOT INTRODUCED. FSAR TABLE 6.3-1 LISTS THE FAILURE MODES AND EFFECTS ANALYSIS (FMEA) FOR SI-129A(B) (2SI-FM317A / 2SI-FM348B). THE FMEA POSTULATES AN OPERATOR ERROR THAT CAUSES THE VALVE TO BE CLOSED RESULTING IN A LOSS OF ONE LPSI TRAIN. THE INHERENT COMPENSATING PROVISION REMAINS THE REDUNDANT PARALLEL LPSI TRAIN WITH THE PROPOSED CHANGE. THE DESIGN MAINTAINS SEPARATION AND REDUNDANCY BY USING A SEPARATE BACKUP AIR SUPPLY SYSTEM FOR EACH

TRAIN. 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: THE CHAPTER 15 ACCIDENTS EVALUATED IN THE UFSAR THAT MAY BE AFFECTED BY A MALFUNCTION OF SI-129A(B) ARE LOCA (FSAR 15.6.3.3), STEAM GENERATOR TUBE RUPTURE (SGTR) (FSAR 15.6.3.2), AND OTHER EVENTS THAT EVENTUALLY ACHIEVE SDC ENTRY CONDITIONS. THE POTENTIAL MALFUNCTIONS INTRODUCED BY MODIFYING SI-129A(B) ARE 1) INADVERTENT CLOSURE, WHICH WOULD PREVENT LPSI FLOW, AND 2) FAILURE TO THROTTLE FOR SDC, WHICH WOULD RESULT IN HAVING TO STEAM LONGER TO REMOVE DECAY HEAT. EC-30976 ADDS AN ALTERNATE METHOD OF CONTROLLING SHUTDOWN COOLING (SDC) FLOW RATE AND COOLDOWN RATE IN THE POSTULATED EVENT OF AN EXTENDED LOSS OF INSTRUMENT AIR WHERE THE SAFEGUARD PUMP ROOMS ARE NOT ACCESSIBLE BECAUSE OF RADIATION DOSE RATES. THE BACKUP AIR SUPPLY ENABLES SDC INITIATION WITHIN THE TIMEFRAME ASSUMED IN THE DOSE ANALYSES. ENGINEERED AND ADMINISTRATIVE CONTROLS ARE DESIGNED TO ENSURE THAT POTENTIAL MALFUNCTIONS ARE NOT INTRODUCED. OPENING OR CLOSING THE SI-129A(B) VALVE IS NOT AN ACCIDENT INITIATOR. THE ALTERNATE AIR MODIFICATION TO SI-129 IS TO ENHANCE MITIGATIVE FUNCTION OF COOLING THE CORE VIA SHUTDOWN COOLING SYSTEM IN EVENT OF LOSS OF INSTRUMENT AIR AND RESTRICTION TO OPERATORS TO ENTER SAFEGUARDS PUMP ROOM DUE TO DOSE CONSEQUENCES WHEN ALIGNING SI-129 MANUALLY; THIS MODIFICATION DOES NOT INTRODUCE ANY FACTORS MAKING EXISTING TRANSIENTS AND ACCIDENTS MORE LIKELY IN THAT THE MODIFICATION IS DESIGNED AND OPERATED CONSISTENT WITH MAINTAINING TWO SEPARATE AND INDEPENDENT SHUTDOWN COOLING TRAINS; CONTROL OF SI-129 IS BY OPERATORS TRAINED IN THE MODIFICATION USING APPROVED PROCEDURES AND LINEUP IS PERIODICALLY MONITORED TO ENSURE ALTERNATE AIR IS NOT IN SERVICE UNTIL REQUIRED.

THE MANUAL OPERATOR ACTION IS EVALUATED AGAINST NRC INFORMATION NOTICE 97-78, NRC REGULATORY ISSUE SUMMARY 2005-20, AND ANSI/ANS-58.8-1994 ON SYSTEM OPERATION POST RECIRCULATION ACTUATION SIGNAL (RAS). THE TEN PRIMARY ATTRIBUTE EVALUATIONS ARE SPECIFICALLY LISTED BELOW.

(1) THE SPECIFIC OPERATOR ACTIONS REQUIRED;

THE OPERATOR ACTIONS REQUIRED TO PLACE A STANDBY TRAIN OF SHUTDOWN COOLING IN SERVICE IS CONTAINED IN PROCEDURE OP-009-005. POST RETURN TO SERVICE ACTIONS FOR EC-30976 ARE ESTABLISHED TO TRACK PROCEDURE CHANGES AND TRAINING NECESSARY TO PROTECT THE ASSUMPTIONS IN THE ANALYSIS. THE SPECIFIC CHANGE IN OPERATOR ACTION IS TO CLOSE SI-129A(B) BY OPERATING A ½" BALL VALVE, IA-51145A / IA-51185B

IN THE -15 VALVE GALLERY AND CONTROLLING TOTAL SDC FLOWRATE USING SDC TEMPERATURE CONTROL VALVES, SI-415A(B), RATHER THAN USING THE SI-129A(B) VALVE HANDWHEEL FROM THE SAFEGUARD PUMP ROOM A(B) TO CONTROL TOTAL SDC FLOWRATE AND ONLY CONTROLLING TEMPERATURE USING SI-415A(B). NEITHER OF THE MANUAL ACTIONS WOULD BE REQUIRED EXCEPT IN THE EVENT OF A LOSS OF INSTRUMENT AIR. THE ORIGINAL MANUAL ACTION PREVIOUSLY EVALUATED WOULD BE AVAILABLE EXCEPT IN THE EVENT THAT THE SAFEGUARD ROOM WOULD BE INACCESSIBLE. MANUAL OPERATOR ACTION IN THE -15 VALVE GALLERY IS ALREADY REQUIRED TO CLOSE CS-117A(B) IN ORDER TO PLACE A STANDBY TRAIN OF SHUTDOWN COOLING IN SERVICE. THEREFORE, THERE IS NO ADVERSE IMPACT.

(2) THE POTENTIALLY HARSH OR INHOSPITABLE ENVIRONMENTAL CONDITIONS EXPECTED;

THE EVENT OF INTEREST IS A LOCA (FSAR 15.6.3.3), STEAM GENERATOR TUBE RUPTURE (SGTR) (FSAR 15.6.3.2), OR OTHER EVENT THAT EVENTUALLY ACHIEVES SDC ENTRY CONDITIONS. THE ENVIRONMENTAL CONDITIONS IN THE SAFEGUARD PUMP ROOM FOR SOME OF THE EVENTS COULD POTENTIALLY BE HARSH OR INHOSPITABLE. THE ENVIRONMENTAL CONDITIONS IN THE -15' VALVE GALLERY HAVE BEEN PREVIOUSLY EVALUATED TO ALLOW CLOSING CS-117A(B) TO ALIGN SDC FOR OPERATION. CALCULATION OSA-RC-CALC-91001 CONCLUDES THAT THE TOTAL DOSE FOR A 10 MINUTE STAY TIME IN THE AREA OF CS-117A AND CS-117B WOULD BE 3.85 R AND 2.68 R, RESPECTIVELY. 10 MINUTES IS IN AGREEMENT WITH THE VALUE IN FSAR TABLE 12.3A-9 AND WOULD BE AMPLE TIME TO CLOSE CS-117A(B) AND TURN IA-51145A / IA-51185B. THE OPERATOR ACTION TO UNLOCK AND TURN THE HANDLE OF IA-51145A / IA-51185B 90 DEGREES IN ORDER TO CLOSE SI-129A(B) COULD REASONABLY BE PERFORMED IN LESS THAN ONE MINUTE WHILE IN THE -15 VALVE GALLERY. THE CALCULATED DOSE RATE WOULD ALLOW UP TO 13 MINUTES STAY TIME PRIOR TO REACHING THE 5 R LIMIT SET FORTH IN FSAR SECTION 12.3A. THEREFORE, THERE IS NO ADVERSE IMPACT.

(3) A GENERAL DISCUSSION OF THE INGRESS/EGRESS PATHS TAKEN BY THE OPERATORS TO ACCOMPLISH FUNCTIONS;

THE MAJORITY OF THE OP-009-005 ACTIONS OCCUR FROM THE CONTROL ROOM. EXISTING ACTIONS INCLUDE MANUAL LOCAL OPERATOR ACTION TO CLOSE CS-117A(B) FROM THE -15 RAB VALVE GALLERY, WHICH IS THE LOCATION OF THE NEW AIR ACCUMULATOR AND THREE WAY BALL VALVE IA-51145A / IA-51185B. EMERGENCY LIGHTING IS AVAILABLE IN ALL LOCATIONS NEEDED AND THE ENVIRONMENT CONDITIONS (2) WILL NOT BE ADVERSE. POST RETURN TO SERVICE ACTIONS FOR EC-30976 ARE ESTABLISHED TO TRACK PROCEDURE CHANGES AND TRAINING NECESSARY TO PROTECT THE ASSUMPTIONS IN THE ANALYSIS. THEREFORE, THERE IS NO ADVERSE IMPACT.

(4) THE PROCEDURAL GUIDANCE FOR REQUIRED ACTIONS;

THE OPERATOR ACTIONS REQUIRED TO PLACE A STANDBY TRAIN OF SHUTDOWN COOLING IN SERVICE IS CONTAINED

IN PROCEDURE OP-009-005. POST RETURN TO SERVICE ACTIONS FOR EC-30976 ARE ESTABLISHED TO TRACK PROCEDURE CHANGES AND TRAINING NECESSARY TO PROTECT THE ASSUMPTIONS IN THE ANALYSIS. THE SPECIFIC CHANGE IN OPERATOR ACTION IS TO CLOSE SI-129A(B) BY OPERATING A ½" BALL VALVE, IA-51145A / IA-51185B IN THE -15 VALVE GALLERY AND CONTROLLING TOTAL SDC FLOWRATE USING SDC TEMPERATURE CONTROL VALVES, SI-415A(B), RATHER THAN USING THE SI-129A(B) VALVE HANDWHEEL FROM THE SAFEGUARD PUMP ROOM A(B) TO CONTROL TOTAL SDC FLOWRATE AND ONLY CONTROLLING TEMPERATURE USING SI-415A(B). NEITHER OF THE MANUAL ACTIONS WOULD BE REQUIRED EXCEPT IN THE EVENT OF A LOSS OF INSTRUMENT AIR. THE ORIGINAL MANUAL ACTION PREVIOUSLY EVALUATED WOULD BE AVAILABLE EXCEPT IN THE EVENT THAT THE SAFEGUARD ROOM WOULD BE INACCESSIBLE. MANUAL OPERATOR ACTION IN THE -15 VALVE GALLERY IS ALREADY REQUIRED TO CLOSE CS-117A(B) IN ORDER TO PLACE A STANDBY TRAIN OF SHUTDOWN COOLING IN SERVICE. THEREFORE, THERE IS NO ADVERSE IMPACT.

(5) THE SPECIFIC OPERATOR TRAINING NECESSARY TO CARRY OUT ACTIONS, INCLUDING ANY OPERATOR QUALIFICATIONS REQUIRED TO CARRY OUT ACTIONS;

PLACING SHUTDOWN COOLING IN SERVICE IS ALREADY A MANUAL OPERATOR ACTION THAT IS TRAINED ON AS PART OF OPERATOR REQUALIFICATION TRAINING. THE OPERATORS DEDICATED TO PERFORMING SPECIFIC ACTIONS FOR A POTENTIAL SHUTDOWN COOLING MALFUNCTION WILL BE BRIEFED AS TO COMMUNICATIONS PROTOCOLS, STANDBY AREAS, TRAVEL ROUTES, TIME LIMITS, THE REQUIRED PROCEDURE STEPS, AND ACTIONS REQUIRED. THEREFORE, THERE IS NO ADVERSE IMPACT.

(6) ANY ADDITIONAL SUPPORT PERSONNEL AND/OR EQUIPMENT REQUIRED BY THE OPERATOR TO CARRY OUT ACTIONS;

THE OPERATIONS PERSONNEL AND EQUIPMENT THAT ADDRESS THE CURRENT OPERATION WILL NOT CHANGE. NO ADDITIONAL SUPPORT PERSONNEL OR EQUIPMENT ARE REQUIRED TO CARRY OUT THE ACTIONS TO PLACE THE STANDBY SHUTDOWN COOLING TRAIN IN SERVICE. THEREFORE, THERE IS NO ADVERSE IMPACT.

(7) A DESCRIPTION OF INFORMATION REQUIRED BY THE CONTROL ROOM STAFF TO DETERMINE WHETHER SUCH OPERATOR ACTION IS REQUIRED, INCLUDING QUALIFIED INSTRUMENTATION USED TO DIAGNOSE THE SITUATION AND TO VERIFY THAT THE REQUIRED ACTION HAS SUCCESSFULLY BEEN TAKEN;

PLACING SHUTDOWN COOLING IN SERVICE IS ALREADY A MANUAL OPERATOR ACTION REQUIRING ENTRY TO THE -15 VALVE GALLERY WHERE THE NEW ACCUMULATOR AND THREE WAY BALL VALVE ARE LOCATED. THE CONTROL ROOM STAFF WOULD HAVE INDICATION OF A LOSS OF INSTRUMENT AIR IN ACCORDANCE WITH PROCEDURE OP-901-511, INSTRUMENT AIR MALFUNCTION. RADIATION MONITORING IS ALREADY PROVIDED FOR THE SAFEGUARD PUMP ROOMS TO INDICATE WHETHER THE SAFEGUARD PUMP ROOMS ARE ACCESSIBLE. POST RETURN TO SERVICE ACTIONS FOR EC-30976 ARE ESTABLISHED TO TRACK PROCEDURE CHANGES AND TRAINING NECESSARY TO

PROTECT THE ASSUMPTIONS IN THE ANALYSIS. THE DETERMINATION OF WHETHER OPERATOR ACTION IS REQUIRED BEGINS WITH THE DETERMINATION OF AN ADVERSE IMPACT ON THE SHUTDOWN COOLING SYSTEM. OFF NORMAL PROCEDURE OP-901-131, SHUTDOWN COOLING MALFUNCTION, WILL BE ENTERED FOR ANY ABNORMAL SHUTDOWN COOLING CONDITIONS. THEREFORE, THERE IS NO ADVERSE IMPACT.

(8) THE ABILITY TO RECOVER FROM CREDIBLE ERRORS IN PERFORMANCE OF MANUAL ACTIONS, AND THE EXPECTED TIME REQUIRED TO MAKE SUCH A RECOVERY;

THE ABILITY TO RECOVER FROM CREDITABLE ERROR WILL BE IMMEDIATELY IDENTIFIED BY THE CONTROL ROOM OPERATORS BASED UPON SYSTEM FLOW AND TEMPERATURE INDICATIONS. THE PLACEMENT OF SHUTDOWN COOLING SYSTEM IS ALREADY A MANUAL OPERATOR ACTION SO THIS CHANGE WILL NOT AFFECT THE CREDIBLE ERRORS THAT ALREADY EXISTED. THEREFORE, THERE IS NO ADVERSE IMPACT.

(9) CONSIDERATION OF THE RISK SIGNIFICANCE OF THE PROPOSED OPERATOR ACTIONS;

THE RISK SIGNIFICANCE FOR PLACING SHUTDOWN COOLING IN SERVICE HAS NOT CHANGED. THE PROCESS OF PLACING THE STANDBY SHUTDOWN COOLING TRAIN IN SERVICE WAS ALREADY A MANUAL OPERATOR ACTION. THEREFORE, THERE IS NO ADVERSE IMPACT.

(10) TIME RESPONSE AS OUTLINED IN ANSI/ANS-58.8-1994, "TIME RESPONSE DESIGN CRITERIA FOR SAFETY-RELATED OPERATOR ACTION";

ANSI/ANS-58.8-1994 PROVIDES TIME REQUIREMENTS FOR DIFFERENT PLANT CONDITIONS AND EXPLICITLY LIMITS ITS INTENDED APPLICATION TO TIME CRITICAL OPERATOR ACTIONS. THIS CHANGE ONLY IMPACTS THE OPERATOR ACTIONS THAT WOULD BE REQUIRED TO INITIATE SHUTDOWN COOLING AFTER AN ACCIDENT THAT RESULTS IN HIGH DOSE RATES IN THE SAFEGUARD PUMP ROOMS ALONG WITH A LOSS OF INSTRUMENT AIR. THE TIME ALLOWED TO INITIATE SHUTDOWN COOLING WOULD BE ON THE ORDER OF SIX TO EIGHT HOURS AFTER THE INDICATION OF THE EVENT. THEREFORE, THE TIME AVAILABLE EXCEEDS THE MINIMUM TIME REQUIREMENTS CONTAINED IN ANSI 58.8 AND THE OPERATOR ACTION IS CREDIBLE.

THE SPECIFIC CHANGE IN OPERATOR ACTION IS TO CLOSE SI-129A(B) BY OPERATING A ½" BALL VALVE, IA-51145A / IA-51185B IN THE -15 VALVE GALLERY AND CONTROLLING TOTAL SDC FLOWRATE USING SDC TEMPERATURE CONTROL VALVES, SI-415A(B), RATHER THAN USING THE SI-129A(B) VALVE HANDWHEEL FROM THE SAFEGUARD PUMP ROOM A(B) TO CONTROL TOTAL SDC FLOWRATE AND ONLY CONTROLLING TEMPERATURE USING SI-415A(B). EC-30976 EVALUTES THAT THE NEW MANUAL ACTION CAN BE ACCOMPLISHED IN LESS TIME THAN LOCAL MANUAL OPERATION VIA HANDWHEEL AND THE AREA OF -15 VALVE GALLERY IS FULLY ACCESSIBLE IN ACCIDENT SCENARIOS OF APPLICABLE SAFETY ANALYSIS.

CALCULATION ECM11-003 IS THE DESIGN BASIS CALCULATION FOR THE BACKUP AIR SUPPLY MISSION TIME AND

DOCUMENTS THE EVALUATION OF THE TIME FOR OPERATOR ACTION. THEREFORE, THERE IS NO ADVERSE IMPACT.

THEREFORE, THE PROPOSED CHANGE DOES NOT 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: THIS CHANGE AFFECTS THE SHUTDOWN COOLING FLOW CONTROL VALVE, SI-129A(B) AND A NEW SAFETY RELATED BACKUP INSTRUMENT AIR SYSTEM. SI-129A(B) IS NORMALLY OPEN TO PROVIDE A FLOWPATH FOR LPSI. A BACKUP AIR SUPPLY CAPABLE OF CLOSING SI-129A(B) VALVE FROM THE -15 VALVE GALLERY FOR SDC OPERATION IS BEING INSTALLED. THE SYSTEM WILL BE CONTROLLED BY A 3-WAY BALL VALVE, WHICH WILL NORMALLY BE LOCKED TO ISOLATE THE SYSTEM, PREVENTING INADVERTENT CLOSURE OF THE SI-129A(B) VALVE. OPENING OR CLOSING THE SI-129A(B) VALVE IS NOT AN ACCIDENT INITIATOR. FSAR TABLE 6.3-1 LISTS THE FAILURE MODES AND EFFECTS ANALYSIS (FMEA) FOR SI-129A(B) (2SI-FM317A / 2SI-FM348B). THE FMEA POSTULATES AN OPERATOR ERROR THAT CAUSES THE VALVE TO BE CLOSED RESULTING IN A LOSS OF ONE LPSI TRAIN. THE INHERENT COMPENSATING PROVISION REMAINS THE REDUNDANT PARALLEL LPSI TRAIN WITH THE PROPOSED CHANGE. THE FMEA ALSO STATES THAT VALVE IS NORMALLY LOCKED OPEN, EXCEPT DURING SDCS OPERATION. THE PROPOSED CHANGE EMPLOYS A LOCKED 3-WAY BALL VALVE SUCH THAT THE EXISTING FMEA REMAINS VALID. THE MODIFICATION IS DESIGNED AND OPERATED CONSISTENT WITH MAINTAINING TWO SEPARATE AND INDEPENDENT SHUTDOWN COOLING TRAINS. THEREFORE, THE PROPOSED CHANGE 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 DESIGN BASIS LIMITS FOR FISSION PRODUCT BARRIERS ARE:

FUEL CLADDING – PEAK LINEAR HEAT RATE = 21 KW/FT, PEAK FUEL CENTERLINE TEMPERATURE = 5080°F (DECREASING BY 58°F PER 10,000 MWD/MTU FOR BURNUP AND ADJUSTING FOR BURNABLE POISONS PER CENPD-382-P-A.), DNBR = 1.24,

RCS BOUNDARY – 2750 PSIA

CONTAINMENT PRESSURE – 44 PSIG

THIS CHANGE AFFECTS THE SHUTDOWN COOLING FLOW CONTROL VALVE, SI-129A(B) AND A NEW SAFETY RELATED BACKUP INSTRUMENT AIR SYSTEM. SI-129A(B) IS NORMALLY OPEN TO PROVIDE A FLOWPATH FOR LPSI. A BACKUP AIR SUPPLY CAPABLE OF CLOSING SI-129A(B) VALVE FROM THE -15 VALVE GALLERY FOR SDC OPERATION IS BEING INSTALLED. THE SYSTEM WILL BE CONTROLLED BY A 3-WAY BALL VALVE, WHICH WILL

NORMALLY BE LOCKED TO ISOLATE THE SYSTEM, PREVENTING INADVERTENT CLOSURE OF THE SI-129A(B) VALVE. OPENING OR CLOSING THE SI-129A(B) VALVE IS NOT AN ACCIDENT INITIATOR. FSAR TABLE 6.3-1 LISTS THE FAILURE MODES AND EFFECTS ANALYSIS (FMEA) FOR SI-129A(B) (2SI-FM317A / 2SI-FM348B). THE FMEA POSTULATES AN OPERATOR ERROR THAT CAUSES THE VALVE TO BE CLOSED RESULTING IN A LOSS OF ONE LPSI TRAIN. THE INHERENT COMPENSATING PROVISION REMAINS THE REDUNDANT PARALLEL LPSI TRAIN WITH THE PROPOSED CHANGE. THIS CHANGE HAS NO IMPACT ON ANY OF THE DESIGN BASIS LIMITS FOR FISSION PRODUCT BARRIERS AS THE CHANGE IS TO MITIGATIVE SYSTEM TO ENHANCE PERFORMANCE OF INITIATING SHUTDOWN COOLING UPON LOSS OF INSTRUMENT AIR IN ACCIDENT RECOVERY. THEREFORE, THERE IS NO IMPACT ON THE DESIGN BASIS LIMITS FOR THE FUEL CLADDING, THE REACTOR COOLANT SYSTEM PRESSURE BOUNDARY, OR THE CONTAINMENT. THEREFORE, THIS CHANGE DOES NOT IMPACT ANY LIMIT FOR A FISSION PRODUCT BARRIER AS DESCRIBED IN THE UFSAR.

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 CALCULATION METHODS USED FOR EVALUATING THE SYSTEM BEHAVIOR AND RESPONSE USING THE ALTERNATE VALVE CONTROL METHOD ARE NOT CHANGED. ANS 5.1-94 METHODOLOGY WAS USED TO DETERMINE DECAY HEAT VS. TIME AFTER SHUTDOWN. THIS IS THE SAME METHODOLOGY USED IN THE EXISTING DESIGN BASIS CALCULATION CN-PS-03-15, SHUTDOWN COOLING SYSTEM PERFORMANCE FOR W3 3716 MWT EPU. THE RESULTS ARE VALIDATED WITH EPU CALCULATION CN-PS-03-15. STER 5.04 HEAT EXCHANGER PERFORMANCE PREDICTION SOFTWARE IS USED TO MODEL THE SDC HEAT EXCHANGER WITH VARIOUS CCW FLOW RATES AND SDC FLOW RATE AT 4100 GPM. THE GEOMETRY INPUTS ARE THE SAME AS USED IN EXISTING DESIGN BASIS CALCULATION MNQ9-1, SDC HX U-VALUE. THE ALLOWABLE ACCUMULATOR LEAKAGE CALCULATIONS USE THE SAME METHODOLOGY AS USED FOR OTHER AIR ACCUMULATORS IN ECM89-089, ALLOWABLE INSTRUMENT AIR ACCUMULATOR LEAK RATE. 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 # 12-2 / Rev. #:** 0

Proposed Change / Document: CR-WF3-2012-3280 documents that valve ACC-126A (CCW 'A' Temperature Control Valve) may not be fully seated (i.e , closed) when required to preserve Wet Cooling Tower (WCT) inventory margin post-accident. This proposed change is to update EP-002-100 to require a manual operator action to close ACC-126A when there is no longer a demand for the valve to be open post-accident in order to prevent excess water inventory loss from the Wet Cooling Tower (WCT) Basins.

Description of Change:

This 50 59 evaluation addresses the potentially adverse impact of the operator action required during the EP-002-100 (TECHNICAL SUPPORT CENTER (TSC) ACTIVATION, OPERATION AND DEACTIVATION) in order to close ACC-126A in order to prevent excess water inventory losses in the event that the actuator's I/P modulator has drifted where it will not adequately close the valve when demanded. The specific action required is to manually close ACCMVAAA126A (ACC HEADER A CCW HX OUTL TEMPERATURE CONTROL) using the handwheel to ensure the valve is adequately closed to maintain adequate Wet Cooling Tower Basin water inventory. The purpose of this 50.59 evaluation is not to justify or evaluate the acceptability of leakage past ACC-126A if not fully seated, but only to evaluate the effects of the Compensatory Measure (i.e., locally closing valve manually after post-accident function is no longer required) on other aspects of the facility (ref EN-OP-104 [6]).

The WCT is designed to operate whenever the heat rejection capacity of the DCT is exceeded. Each WCT is capable of dissipating the maximum heat load from the CCW Heat Exchanger plus an Essential Chiller. The volume of water stored in each basin is sufficient to bring the plant to a safe shutdown under all design basis accident conditions. During a large break LOCA, the volume of water in one wet cooling tower basin is sufficient to provide UHS operation for essential loads without makeup. When adding the non-essential load of fuel pool cooling, approximately 113,807 gallons of makeup is required to the WCT basin if only one UHS train is available. Adequate makeup can be supplied to a WCT basin by gravity feeding from the alternate WCT basin. The WCT basins can be manually interconnected through a Seismic Category I line to take advantage of the water volume in both basins.

ACC-126Ais a 12" air operated temperature control butterfly valve in the ACC return to the wet cooling towers and opens as required to cool CCW and transfer heat load to Wet Tower during normal operation and accident conditions when the Dry Cooling Tower (DCT) alone cannot remove all the heat load. This valve is normally closed and must open when accident and/or environmental conditions cause the heat rejection capabilities of the DCT to be exceeded. When in automatic, this valve opens to maintain CCW temperature below 95°F during normal operation and at 115°F post-accident. This valve must throttle to conserve Wet Tower inventory during accident events as analyzed in calculation MNQ9-9. When a Safety Injection Actuation Signal (SIAS) occurs, the setpoint for ACC to supply the CCW heat exchanger is automatically raised from 95°F to 115°F provided wet tower basin temperature is >74°F. As decay heat lowers following an accident, the demand for ACCW flow through the CCW Heat Exchanger will lower The current design analysis concludes that the CCW Heat Exchanger will no longer be needed to transfer heat from CCW after 4 days post accident.

The analysis assumes that after day eight the Essential Chiller is placed back on CCW when temperatures are 110 F or less and the WCT is removed from service

Technical Specification 3.7.4 requires Wet Cooling Tower basin level to be maintained above 97% indicated level This ensures that there is adequate water inventory to bring the plant to safe shutdown under all design basis accident conditions accounting for evaporation and drift losses from operating the wet cooling towers. This compensatory measure is to ensure the WCT basin water inventory is maintained to perform its safety function.

The ACCW System consists of two independent, 100% capacity trains Each train contains an ACCW Pump, a CCW Heat Exchanger a modulating temperature control valve and an evaporative, wet-type, mechanical draft

[1] 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.

cooling tower. The ACCW System transports heat from the CCW Heat Exchangers to the Wet Cooling Towers when required. Cycling Wet Cooling Tower fans enhances the heat rejection capability When necessary, the ACCW System also removes heat from the safety related Essential Chillers. The ACCW System is required to operate whenever the heat rejection capacity of the CCW System Dry Cooling Towers is exceeded (LOCA or MSLB conditions) and any time the CCW System Dry Cooling Towers cannot sufficiently reject heat load due to outside ambient conditions ACCW A flow is modulated by an air-operated valve, ACC-126A, to control CCW System supply header temperature at setpoint. At the WCTs, flow goes out spray header nozzles and falls into the basins. Eight WCT Fans draw air upward through the falling droplets, evaporating a small portion of each drop. The evaporation process leaves the remainder of each droplet several degrees cooler in temperature. The cooled water droplets collect in the basins for reuse by the ACCW Pumps

The Operability Evaluation for CR-WF3-2012-3280 credits a compensatory measure for local operator action to manually close ACC-126A several days into a design basis event, when the CCW Heat Exchanger is no longer required to maintain CCW supply temperature at 115°F, in order to prevent excess WCT basin inventory losses.

If a new operator action must be performed in order for a system to be capable of performing its specified safety function, then it must be evaluated with respect to the guidance presented in NRC INFORMATION NOTICE 97-78 [Reference 2], NRC REGULATORY ISSUE SUMMARY 2005-20 [Reference 3], and NRC Inspection Manual part 9900 [Reference 4].

NRC INFORMATION NOTICE 97-78 [Reference 2] lists the following requirements for crediting an operator action. The original design of nuclear power plant safety systems and their ability to respond to design-basis accidents were described in licensees' FSARs and were reviewed and approved by the NRC. Most safety systems were designed to rely on automatic system actuation to ensure that the safety systems were capable of carrying out their intended functions. In a few cases, limited operator actions, when appropriately justified, were approved. Proposed changes that substitute manual action for automatic system actuation or modify existing operator actions, including operator response times, previously reviewed and approved during the original licensing review of the plant will, in all likelihood, raise the possibility of a USQ. Such changes must be evaluated under the criteria of 10 CFR 50.59 to determine whether a USQ is involved and whether NRC review and approval is required before implementation. A licensee may not make such changes before it receives approval from the NRC when the change, test, or experiment may (1) increase the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously analyzed in the FSAR, (2) create the possibility of an accident or a malfunction of a different type than any previously evaluated in the FSAR, or (3) reduce the margin of safety as defined in the basis for any TS. In the NRC staff's experience, many of the changes of the type described above proposed by licensees do involve a USQ.

NRC INFORMATION NOTICE 97-78 also lists specific requirements the NRC will use to review new operator actions. Based on these guidelines, the NRC's reviews of licensees' analyses typically include, but are not limited to, (1) the specific operator actions required; (2) the potentially harsh or inhospitable environmental conditions expected; (3) a general discussion of the ingress/egress paths taken by the operators to accomplish functions; (4) the procedural guidance for required actions; (5) the specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions; (6) any additional support personnel and/or equipment required by the operator to carry out actions; (7) a description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken (8) the ability to recover from credible errors in performance of manual actions and the expected time required to make such a recovery, and (9) consideration of the risk significance of the proposed operator actions

NRC REGULATORY ISSUE SUMMARY 2005-20 [Reference 3] issued a new version of NRC Inspection Manual Technical Guidance Part 9900 ITSB [Reference 4]. The NRC Inspection Manual [Reference 4] lists the following requirements for crediting an operator action

For situations where substitution of manual action for automatic action is proposed for an operability determination, the evaluation of manual action must focus on the physical differences between automatic and manual action and the ability of the manual action to accomplish the specified safety function or functions The physical differences to be considered include the ability to recognize input signals for action, ready access to or recognition of setpoints, design nuances that may complicate subsequent manual operation (such as auto-reset, repositioning on temperature or pressure), timing required for automatic action, minimum staffing requirements and emergency

operating procedures written for the automatic mode of operation. The licensee should have written procedures in place and personnel should be trained on the procedures before any manual action is substituted for the loss of an automatic action.

The assignment of a dedicated operator for a manual action requires written procedures and full consideration of all pertinent differences. The consideration of a manual action in remote areas must include the abilities of the assigned personnel and how much time is needed to reach the area, training of personnel to accomplish the task, and occupational hazards such as radiation, temperature, chemical, sound, or visibility hazards. One reasonable test of the reliability and effectiveness of a manual action may be the approval of the manual action for the same function at a similar facility.

The manual operator action to close ACC-126A locally was evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 and is addressed in this 50.59 evaluation.

References

1. Technical Specification Amendment 230.
2. NRC INFORMATION NOTICE 97-78, CREDITING OF OPERATOR ACTIONS IN PLACE OF AUTOMATIC ACTIONS AND MODIFICATIONS OF OPERATOR ACTIONS, INCLUDING RESPONSE TIMES, October 23, 1997.
3. NRC REGULATORY ISSUE SUMMARY 2005-20 REV. 1, REVISION TO NRC INSPECTION MANUAL PART 9900 TECHNICAL GUIDANCE, OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS FOR RESOLUTION OF DEGRADED OR NONCONFORMING CONDITIONS ADVERSE TO QUALITY OR SAFETY, April 16, 2008.
4. NRC Inspection Manual Technical Guidance Part 9900 ITSB, OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS FOR RESOLUTION OF DEGRADED OR NONCONFORMING CONDITIONS ADVERSE TO QUALITY OR SAFETY
5. ANSI/ANS-58.8-1994, American National Standard Time Response Design Criteria for Safety Related Operator Actions, August 23, 1994.
6. NRC GENERIC LETTER 98-02, LOSS OF REACTOR COOLANT INVENTORY AND ASSOCIATED POTENTIAL FOR LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION, May 28, 1998.
7. MNQ9-9
8. CR-WF3-2012-3154
9. W3-DBD-4, Component Cooling Water / Auxiliary Component Cooling Water Design Basis Document
10. CR-WF3-2009-630
11. EC-14006, ACC-126A(B) Manual Handwheel Safety Function

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: Dale Gallodoro / Entergy / DE-ME [signature] 7-10-2012
Name (print) / Signature / Company / Department / Date

Co-Preparer: Marc McCloskey / Entergy / DE-ME [signature] 7-10-12
Name (print) / Signature / Company / Department / Date

Reviewer: Bill Hardin / Entergy / Licensing / [signature] 7/10/12
Name (print) / Signature / Company / Department / Date

Reviewer: Pete McKenna / Entergy / System Engineering / [signature] 7/10/12
Name (print) / Signature / Company / Department / Date

OSRC: David Hamilton / per Telecon / 7-10-12 [signature] for D Hamilton per telecon.
Chairman's Name (print) / Signature / Date

OSRC Meeting # W3 12-11

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:

   The proposed change requires manual closure of ACC-126A following a design basis accident at a time when ACCW flow is not needed for CCW temperature control. ACC-126A valve is not an accident initiator. UFSAR Chapter 6 and 15 were reviewed to identify which accidents previously evaluated could be initiated or caused by the proposed change. Manual operation of ACC-126A is only required when the CCW Hx is no longer needed to remove heat from CCW in order to ensure that adequate water inventory remains in the wet cooling tower basin to continue to cool the load from the essential chillers.

   The proposed change also does not create any new system interactions and has no impact on operation or function of any system or equipment that in any way could cause an accident. Therefore, the proposed change does not result in an 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 proposed change does not increase the likelihood of occurrence of a malfunction of a SSC important to safety. The change will not impact the ability of ACC-126A to modulate to control CCW Hx outlet temperature. The proposed action requires manual closure of ACC-126A when heat removal by the CCW Hx is no longer required. The manual operator action enhances the operability of the ACC-126A valve and prevents potential malfunction of this valve by fully closing the valve to prevent potential loss of water inventory. The operator action put in place by this change is evaluated in accordance with NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 under Question 6.

   W3-DBD-4, as updated by EC-14006, clarifies the safety function of the ACC-126A manual handwheel. CR-WF3-2009-0630 documents the verification that only safety related qualified parts are currently installed and authorized to be installed in the ACC-126A manual actuator application. The indicator, CC ITIC7070A, is safety related as verified by review of the Indus Asset Suite Equipment Database (EDB).

   As part of the manual closure of ACC-126A the valve will first briefly be taken to 20% open prior to closure. This expected open period is expected to be less than 5 minutes and will have negligible impact on Wet Cooling Tower basin inventory and maintaining CCW temperature.

   The proposed change also does not create any new system interactions and has no impact on operation or function of any system or equipment that in any way could cause increase the likelihood of occurrence of malfunction of the SSC. Therefore, the proposed change does not result in an increase in likelihood of occurrence of 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:

The subject change does not increase the consequences of an accident previously evaluated in the UFSAR. The potential impact of ACC-126A not closing adequately is that WCT basin inventory would be depleted prior to completing its safety related function. The Wet Cooling Tower continues to remove heat from the Essential Chillers for seven (7) days after an accident. After approximately four (4) days ACCW no longer needs to remove heat from the CCW Hx and is only needed for removing heat from the Essential Chillers. Therefore, the proposed action will prevent the potential malfunction of ACC-126A not fully closing when ACCW flow is no longer needed and enhances it's capability to perform its safety function. The operator action put in place by this change is evaluated in accordance with NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 under Question 6

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:

The proposed change does not result in more than a minimal increase in the consequence of the malfunction of an SSC previously evaluated in the UFSAR. The consequences of a malfunction of ACC-126A are bounded by those scenarios evaluated in the UFSAR. The proposed change does not impact the safety function of ACC-126A; therefore, the consequence of a malfunction of ACC-126A is not altered by this change. This change is to further limit the potential malfunction of ACC-126A and is evaluated in accordance with NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 under Question 6.

The proposed change also does not create any new system interactions and has no impact on operation or function of any system or equipment that in any way would impact their required safety function 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:

The Auxiliary Component Cooling Water system is an accident mitigating system and is not an UFSAR Chapter 6 or 15 accident initiator. The proposed manual action is only required to prevent excess water inventory losses in the wet cooling tower basin when the CCW Hx is no longer required to remove heat from the CCW system. The valve correctly works to modulate flow to ensure CCW HX outlet temperature is maintained at its design setpoint, except when the demand is for the valve to be fully closed. The potential impact of the valve not adequately closing would be on the accident mitigating system and the ability of the WCTs ability to remove the heat from the essential chillers. The proposed change has no impact on the function of ACC-126A and ACC-126A is not an accident initiator. Therefore, the change does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR.

The proposed change also does not create any new system interactions. Thus, this proposed change does not 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 potential failure mechanism is that ACC-126A would not go closed once CCW demand is no longer required and that the WCT A basin water volume would not be maintained. This change will substitute automatic closure with manual operator action. Based on the following discussion, the safety function will be maintained:

The manual operator action is evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 on system operation. The ten primary attribute evaluations are specifically listed below.

(1) The specific operator actions required;
The operator actions required to operate ACC-126A are already contained in OP-002-001 section 8.4. The specific action required is to manually locally close the valve using the handwheel. EP-002-100 already contains guidance on monitoring WCT basin inventory after an accident. Guidance is being added EP-002-100 by this change in order to ensure the valve is manually closed using the handwheel to prevent excess water inventory losses when there is no longer a need for the ACCW system to remove heat from the CCW system. EP-002-100 is specifically updated to open ACC-126A to 20% and then take to close to ensure that the valve will be fully seated. This action will also be peer checked. No adverse impact

(2) The potentially harsh or inhospitable environmental conditions expected
Per W3-DBD-14 valve ACC-126A is located in the CCW heat exchanger A room approximately 12 feet off of the floor. Radiation levels are approximately 1 Rem/hr one day post-accident (ref FSAR figure 12.3A-7) for normal ingress/egress routes. ECS09-005 rev. 1 identifies the maximum dose rate for ACC-126A as 1.77 Rem/hr 8 hours after the accident and 0.987 Rem/hr 24 hours after an accident. These dose rates permit limited occupancy. The maximum dose estimate of performing this activity is approximately 250 mrem per operator in order to perform the action in the 15 minutes estimated when operators are dispatched to close the valve, which is acceptable. Therefore the environment is acceptable.

(3) A general discussion of the ingress/egress paths taken by the operators to accomplish functions;
Operator actions will be required in the CCW Hx A room. All Nuclear Auxiliary Operators carry

flashlights which will provide adequate lighting for ingress and egress to the valve. The environment conditions (2) will not be adverse. There are numerous routes that are available for ingress and egress to ACC-126A (see dose discussion in (2)). Temporary scaffolding is currently erected for access to ACC-126A. A restriction is placed on this scaffolding to ensure it is not removed prior to resolution of CR-WF3-2012-3280. No adverse impact.

(4) The procedural guidance for required actions;
The operator actions required to operate ACC-126A are already contained in OP-002-001 section 8.4. Guidance is being added to the emergency procedures (EP-002-100) by this change in order to ensure the valve is manually closed to prevent excess water inventory losses when there is no longer a need for the ACCW system to remove heat from the CCW system. No adverse impact.

(5) The specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions;
Manually operating ACC-126A is already an operator action that is trained on as part of operator requalification training. Local manual operation of ACC-126A is part of NAO requal training (WPPT-AOA-PP000 Rev 5) and was recently covered in Cycle 2 of 2012 as a dynamic learning activity (WDLA-AOR-122DLA). The operators dedicated to performing specific actions for a potential need to manually close ACC-126A will be briefed on the required procedure steps and actions required. TSC operators and engineers will also be briefed on the required actions.

(6) Any additional support personnel and/or equipment required by the operator to carry out actions;
This action will be required several days into the event and the emergency response organization activated. No additional support personnel or equipment are required to carry out the actions to manually operate ACC-126. No adverse impact.

(7) A description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken;
Operators will be able to see when there is no longer a demand for ACC flow to the CCW Hx. Instrument CC ITIC7070A in the control room shows the flow demand for the CCW Hx. When there is no longer a sustained flow demand and the TSC has determined that heat removal from the CCW Hx is no longer required an operator will be dispatched to manually close the handwheel on ACC-126A. Operators will be able to verify the required action was taken successfully by ensuring ACC-126A is closed in accordance with the procedural guidance added to EP-002-100 which has repeatedly been shown to close the valve to the correct position to ensure that WCT basin water inventory is maintained. The procedural guidance added instructs operators to take ACC-126A to 20% open and then to closed in order to get a good seal. Four different operators independently and successfully performed the proposed procedural steps, each within approximately 15 minutes. The results were that each operator did not have difficulty understanding the instructions or performing the instructions and that a good seal was achieved as indicated by greater than 20 psig high point pressure which indicates less than 110 gpm leakage and adequate water inventory can be maintained. This was performed with the Jockey Pump for the system in operation. In an accident condition the main pump would be operating, which would result in potentially higher operating pressure. However differential pressure across ACC-126A would be minimized since back pressure would be present with the Essential Chillers aligned. The procedural guidance includes the requirement for peer checking. Therefore the procedural guidance can be verified as successfully completed and there is reasonable assurance that the manual closure will result in adequate closure to prevent excess water inventory losses.

(8) The ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery
Potential credible errors are evaluated as follows:
1) Failure to recognize the need for action – The failure to recognize the need for action is not considered to be credible since the EP-002-100 procedure directs monitoring for proper UHS component operation well before this action would be required for the worst case water consumption scenario (36 hours vs 4 days). Adequate Engineering and Operations resources

exist in the TSC to coordinate with the control room operators to monitor for ACC-126A performance. If normal makeup to the WCT Basins was unavailable, UHS Inventory would be a high priority of the E-Plan and Operations and would result in appropriate monitoring and contingency planning.

2) Failure to close ACC-126A could be caused by either a human performance error or equipment malfunction. The procedure guidance is specific to ACC-126A and the valve is clearly labeled, assuring the correct component is manipulated. Additionally, EP-002-100 has been updated to specify the action to close ACC-126A be performed by two operators, a performer and peer checker, to minimize any potential for human error. Multiple operators demonstrated the capability to achieve closure with the manual handwheel which was verified by ACCW high point header pressure greater than 20 psig. Equipment malfunction of ACC-126A would constitute an additional single failure, which is beyond design basis (since the wet cooling tower water consumption has already assumed the most bounding single failure). Manual operation of ACC-126A is specified in the IST Program to be tested every 2 years and was last tested on 2/3/11 per OP-903-118 (ref WO# 257038). Additionally, Operations verified the proposed action is implementable and repeatable. Should a malfunction of ACC-126A occur, preventing closure, another manual valve exists downstream (ACC-127A) which could be closed.

(9) Consideration of the risk significance of the proposed operator actions;
There is no impact on risk from performing this action. This manual action to ensure the design basis is maintained.

(10) Time response as outlined in ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-Related Operator Action";
ANSI/ANS-58.8-1994 provides time requirements for different accident scenarios. The proposed manual action from this change is not required until at least 36 hours into an accident. The proposed manual action is needed when there is the largest heat load to be removed which is during a LOCA scenario. Using the ANSI/ANS-58.8-1994 guidance 20 minutes would be required for identification and diagnosis with an additional 1 minute per each required operator action. 30 minutes is also listed in ANSI/ANS-58.8-1 for operator actions required outside the control room. Therefore at least 51 minutes is required before the appropriate action can be performed. Since this action is not required until at least 36 hours into the accident there is adequate time to perform this function.

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 proposed change ensures the Ultimate Heat Sink remains capable of performing it specified function. Therefore the proposed change would not impact the current design basis limits for fission product barriers such as fuel rod cladding, RCS pressure boundary and containment building. Therefore, the proposed change 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:

The existing UFSAR evaluations pertaining to the Auxiliary Component Cooling Water and the Ultimate Heat Sink have not changed. There are no new analysis being performed for 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. #:** 2010-09 / 1

Proposed Change / Document: Miscellaneous Rupture Restraint Modifications for the SG/RVCH Replacement / EC8439

Description of Change:

The Replacement Steam Generators (RSGs) being substituted for the Original Steam Generators (OSGs) as part of the Waterford 3 (W3) SG/RVCH Replacement Project have a larger lower shell diameter and new inspection port and handhole locations which differ from those of the OSGs. To accommodate the RSGs and to obtain clearances needed to operate equipment used in performing the changeout of the steam generators, interference portions of the following pipe rupture restraints will be permanently removed:

- R-RC-38-R-5
- R-RC-44RL1-R-16
- R-RC-45RL2-R-4

In addition, depending on clearances between the RSG lower shell / RSG insulation rings and R-RC-39-R-14, which will only be known for certain following RSG installation, it may be necessary to permanently remove portions of R-RC-39-R-14.

These modifications to rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 are considered adverse due to their rendering these SSCs not capable of performing previously assumed design functions, and therefore, these are being reviewed under a 10 CFR 50.59 Evaluation.

Summary of Evaluation:

Applying the guidance of NRC Generic Letter 87-11, EC8439 determined that rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14, which were originally installed to mitigate the effects of arbitrary intermediate pipe breaks, are no longer required to function as piping rupture restraints. Following removal of the interference portions of these rupture restraints, their remaining structural members will retain their required seismic qualification. Since such pipe breaks are deemed to be no longer credible and included in the plant design basis, deletion of portions of these rupture restraints does not result in unacceptable effects to accident frequencies, accident consequences, and accident mitigation factors evaluated under 10 CFR 50.59. No new accidents or equipment malfunctions are created. A design basis for a fission product barrier as described in the UFSAR is not exceeded or altered. The guidance of NRC Generic Letter 87-11 is an NRC-approved methodology that is described in UFSAR Section 3.6.2.1.1.2, and therefore its use 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 analysis.

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

[1] 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.

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

Sid Munshi/ /SMI/Civil/8-15-2012

Preparer: Mark D. Geraldi / / SGT / Licensing / 14-Aug-12
Name (print) / Signature / Company / Department / Date

Reviewer: STEVE A. BENNETT / / EOI (EASON) / RSG LIC / 8-15-12
Name (print) / Signature / Company / Department / Date

OSRC: William McKinney 9-12-12
Chairman's Name (print) / Signature / Date

W3 12-17
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:

   Rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 themselves are not accident initiation mechanisms for the FSAR Chapter 6 and 15 analyses (i.e., a Loss of Coolant Accident (LOCA – UFSAR Section 15.6.3.3). Portions of these restraints which remain after implementation of EC8439 will retain their seismic qualification. Therefore, there is no 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:

   Rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 were originally installed to mitigate the effects of arbitrary pipe breaks in accordance with NRC Branch Technical Position (BTP) MEB 3-1 of the Standard Review Plan (SRP). NRC Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements" revised MEB 3-1, Section 3.6.2, to eliminate all dynamic effects (missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, compartment, subcompartment and cavity pressurization and decompression waves within the ruptured pipe) resulting from arbitrary intermediate pipe ruptures provided that stress values in the piping are below the specified pipe break threshold stress level. Waterford 3 incorporation of the criteria of NRC Generic Letter 87-11 is described in UFSAR Section 3.6.2.1.1.2.

   As part of EC8464, *Piping/Structural Component and Documentation Impact Review*, Waterford calculations (Ref. Calculation Nos. IM1020, IM1024, IM1131, IM1135) are performed, which determine stress levels at postulated pipe break locations. These calculations determine that for the pipe break locations that originally required rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14, the resultant stress values are less than the NRC Generic Letter 87-11 criteria for their inclusion as potential pipe break locations. Accordingly, with these pipe breaks no longer considered credible, removal of interfering portions of rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 is permitted without impacting SSC malfunction assumptions by implementation of EC8439. Deletion of the function of rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 has no impact on the structural integrity of attached SSCs. Following removal of the interference portions of these rupture restraints, their remaining structural members will retain their required seismic qualification.

   Therefore, for the case of deleting the functions of rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 the proposed change does not 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.

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

BASIS:

Based on the acceptable pipe stresses and the elimination of arbitrary intermediate pipe breaks, deletion of portions of rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, and R-RC-45RL2-R-4, and the potential deletion of structural portions of R-RC-39-R-14 do not reduce the ability of the associated safety injection and shutdown cooling piping to meet its design function for protection against credible dynamic effects. As a result, there is no effect upon the radiological releases postulated to occur for the LOCA caused by the rupture of the affected piping. There is no impact to the pipe failure assumptions for the piping serviced by these rupture restraints as a result of implementing EC8439. Such piping failures and their corresponding dose consequences are bounded by the LOCA dose consequences that are performed for the limiting primary side pipe break described in UFSAR Section 15.6.3.3. 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:

Rupture restraints R-RC-38-R-5 R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 were originally designed to contain the pipe whip effects of a rupture of safety injection and shutdown cooling at arbitrary intermediate pipe break locations. With the application of NRC Generic Letter 87-11, these pipe ruptures are no longer credible. The deletion of portions of these restraints is therefore permitted, since with application of the generic letter these SSCs are no longer required to perform a design basis function to restrain this piping, and are no longer considered SSCs whose failure characteristics could affect dose consequences. The stress values of the associated piping are acceptably low to prevent pipe rupture. 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:

Rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 were originally designed to confine the damage effects or arbitrary intermediate pipe breaks. There are no new accident initiators created by the elimination of these restraint functions. The ruptures of piping serviced by these restraints have been concluded to be not credible, and therefore the possibility for an accident of a different type than any previously evaluated in the UFSAR (i.e., a simultaneous primary and secondary piping break) is not created by the proposed change.

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:

By eliminating arbitrary intermediate breaks associated with rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14 the effects of a pipe breaks at these locations are no longer considered credible. There were no new failure modes created by the modification of these restraints. Therefore, with respect to these four (4) rupture restraints, 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 (i.e., a simultaneous primary and secondary piping break) is not created by the proposed change.

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 in response to this criterion are fuel cladding integrity, RCS pressure boundary integrity, and containment boundary integrity. The only boundary that could be impacted by the proposed change is associated with RCS integrity. NEI 96-07, Revision 1, *Guidelines for 10 CFR 50.59 Implementation*, defines such RCS design basis limits for a fission product barrier as those pertaining to RCS pressure, stresses, and heat-up/cool-down rates. Deleting the pipe rupture restraint function of rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, and R-RC-45RL2-R-4, and the potential deletion of the pipe rupture restraint function of rupture restraint R-RC-39-R-14 have no impact on these design basis limits for the RCS pressure boundary. Therefore, the proposed change does not involve or 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:

The methodology used in evaluating whip restraint loads by EC8439 is consistent with and does not revise or replace UFSAR-described evaluation methodologies for rupture restraints R-RC-38-R-5, R-RC-44RL1-R-16, R-RC-45RL2-R-4, and R-RC-39-R-14. The FSAR established licensing basis approaches for compliance with BTP MEB 3-1 under NRC Generic Letter 87-11 are also not changed. The guidance of NRC Generic Letter 87-11 is an NRC-approved methodology that is described in UFSAR Section 3.6.2.1.1.2, and therefore its use 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 analysis.

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. #:** 2012-03

Proposed Change / Document: ECN40132 (EC8458) Incorporation of RSG Design Document Updates and Review of Design Basis Methodologies for RSG Design Report

Description of Change:

Section 5, *Computer Codes*, of the WCAP-17066-P (Reference 1) identifies five computer programs that were used for evaluating the structural integrity of the Waterford 3 replacement steam generators (RSGs). These codes and their application are:

- ANSYS is used to develop structural heat transfer and stress analysis models for the RSG components, and to derive the temperature profiles and stress states under the combined loadings
- WECAN is used to provide only stiffness matrices for use in the FLOVIB and FASVIB programs which are applied in the flow-induced vibration analyses
- WESTEMS™ – is a post-processor, able to develop transfer function representation of transient stress states based on finite element results derived from ANSYS input
- TRANFLOW - solves the mass, energy, and momentum conservation equations for transient thermal/hydraulic phenomena using a fully implicit backward differencing technique.
- STRIFE is a post-processor providing fatigue analysis based on the ANSYS model database and results

The use of ANSYS has already been addressed in a previous change from EC8458 under a 10CFR50.59 Evaluation. The WECAN and STRIFE codes did not require Updated Final Safety Analysis Report (UFSAR) revision and screened out. The use of WESTEMS™ and TRANFLOW screened in and are being evaluated under this 50.59 Evaluation as a potential change in methods of evaluation.

Summary of Evaluation:

WESTEMS™ was used for the fatigue analysis of the RSG 8x6 inch cone handhole components and 4x3 inch cone inspection port components, both of which are secondary side components. The NRC has approved the use of WESTEMS™ in fatigue applications similar to that applied for Waterford 3. Two issues were identified by the NRC during their review of WESTEMS™ which were addressed in NRC RIS 2011-14. These concerns either did not apply to the Waterford 3 analyses or were satisfactorily resolved for Waterford 3's application. WESTEMS™, as applied on the Waterford 3 RSGs, is consistent with the guidance of NEI 96-07, Section 4.3.8.2 for being an NRC approved methodology and does not constitute a departure from a method as described in the Waterford UFSAR..

TRANFLOW was used for the steamline break accident, determining the pressure drops across SG secondary side internal components in a similar manner that CEFLASH-4A was used as discussed in UFSAR Section 3.9.1.2.2.1.28. TRANFLOW and CEFLASH-4B have been successfully used to evaluate the Westinghouse and Combustion Engineering (CE) fleet of steam generators, respectively. For analyzing blowdown loads, CEFLASH-4A was modified to produce a code version designated as CEFLASH-4B. Even though TRANFLOW has not been approved by the NRC for determining loads on SG internals during a steamline break, it has been shown to produce results comparable to NRC approved codes that were used for that purpose. TRANFLOW results compare favorably with CEFLASH 4A and 4B. TRANFLOW has been developed and maintained under the Westinghouse Quality Assurance (QA) Program. TRANFLOW, as used on Waterford 3, meets the criteria of NEI 96-07, Section 4.3.8.1 that states that changes to elements of analysis methods that yield conservative results, or results that are essentially the same, would not be departures from approved methods.

[1] 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.

References:

1 WCAP-17066-P, Revision 1, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Design Report", October 2012.

2 "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station, Docket Numbers 50-272 and 50-311," U. S Nuclear Regulatory Commission, March 2011

3. Regulatory Information Summary (RIS) 2011-14, Metal Fatigue Analysis Performed by Computer Software, December 29, 2011

4 CN-NCE-W3RSG-11, "Waterford 3 Model Delta 110 Replacement Steam Generator 8 x 6 inch Cone Handhole Analysis," Revision 0, September 2010

5. CN-NCE-W3RSG-12, "Waterford 3 Model Delta 110 Replacement Steam Generator 4 x 3 Inch Cone Inspection Port Analysis," Revision 0, September 2010

6. AP1000 DCD SER Section 3.0, "Design of Structures, Components, Equipment and Systems"

7. Westinghouse LTR-NCE-05-145, "TRANFLOW Computer Code Comparison to CEFLASH-4B Analysis of the Watts Bar Replacement Steam Generator during a Feedwater Line Break", October 14, 2005.

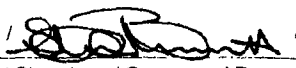
8. CENPD-252-P-A, "Blowdown Analysis Method. Method for the Analysis of Blowdown Induced Forces in a Reactor Vessel," July 1979

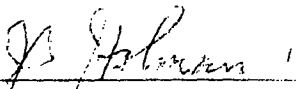
9 Westinghouse LTR-NCE-04-28, Rev. 1, "Position Paper on the Use of the TRANFLO/TRANFLOW Computer Program in Steam Generator Design Analysis," May 13, 2004.

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: Steve Bennett /  / EOI (Iepson) / SGR Licensing / 11-28-2012
Name (print) / Signature / Company / Department / Date

Reviewer: Jerry Holman/  / EOI (SMI) / RSG Eng. / 11/28/12
Name (print) / Signature / Company / Department / Date

OSRC: / Keith Nichols / 12/4/12
Chairman's Name (print) / Signature / Date

W3 12-21
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: N/A

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: N/A

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

   BASIS: N/A

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: N/A

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

   BASIS: N/A

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: N/A

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

   BASIS: N/A

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:

WESTEMS™

WESTEMS™ performs ASME design stress and fatigue analyses using NB-3200 or NB-3600 criteria. WESTEMS is a post-processor, able to develop transient stress states based on transfer function representation of finite element results, derived from ANSYS input. WESTEMS is capable of performing heat transfer and structural analyses covering stress and fatigue evaluations. WESTEMS™ has been used by Westinghouse in stress and fatigue in many nuclear design applications. WESTEMS™ was used for the fatigue analysis of the RSG 8x6 inch cone handhole components and 4x3 inch cone inspection port components, both of which are secondary side components.

The NRC has approved the use of WESTEMS™ in fatigue applications similar to that applied for Waterford 3. In the safety evaluation for the Salem license renewal application (Reference 2) the NRC specifically approves the use of WESTEMS™ for fatigue evaluations based on NB-3200 and did not identify any additional issues concerning the code, except for the recommendation to explicitly document any user adjustment of stress peak and valley results in the final fatigue analysis calculation notes.

However, during NRC review of WESTEMS™ for AP1000 piping fatigue analyses, the NRC raised concerns on its use which were addressed in NRC RIS 2011-14 (Reference 3). Two issues were identified regarding the use of WESTEMS™:

1. The possible misappropriate selection of user-defined option settings in the NB-3600 module of WESTEMS™ that could lead to erroneous moment inputs.
2. The ability of users to manually edit the peak and valley critical times selected automatically by the WESTEMS™ peak and valley selection algorithm, which could possibly result in undocumented, non-conservative results.

Regarding Item 1, the NB-3600 module is used for ASME Section III piping fatigue analyses and, thus, is not applicable to analysis of the Waterford 3 RSGs. Regarding Item 2, the selection of critical times in the RSG handhole and inspection port analyses was performed automatically by the peak and valley selection algorithms in WESTEMS™, and results were reviewed to ensure conservative fatigue results. Calculation notes CN-NCE-W3RSG-11, Rev. 0 (Reference 4) and CN-NCE-W3RSG-12, Rev. 0 (Reference 5) each contain a statement that the automatic selection feature was used and that peak editing was not applied.

NRC identification of the above two issues arose during the NRC's review of the Westinghouse AP1000 Design Control Document (Reference 6). As documented therein, the staff conducted an audit of WESTEMS™ at Westinghouse headquarters and in the Westinghouse Rockville, MD offices. The staff discussed with Westinghouse the theoretical background, formulation, validation methods, and benchmarking pertaining to WESTEMS™. The only issues arising from the review and audit are the two identified in RIS 2011-14, neither of which is applicable to Waterford 3 RSG analyses.

The most conservative interpretation of the general description of the Principal Stress Program, BC 10210, in UFSAR Section 3.9.1.2.2.1.22 is that its use encompassed the purposes for which WESTEMS™ was used for the Waterford 3 RSG analyses. Given that selection of critical times in the RSG handhole and inspection port analyses was performed automatically by the WESTEMS™ peak and valley selection algorithms, WESTEMS™ did not employ either of the only two aspects of the code that the NRC considered to be unacceptable. Based upon the NRC approval for the use of WESTEMS™ for ASME Code NB-3200 fatigue analysis, as was used in the application on the Waterford 3 RSGs, this methodology is consistent with the guidance of NEI 96-07, Section 4.3.8.2 for being an NRC approved methodology and does not constitute a departure from a method as described in the Waterford UFSAR..

TRANFLOW

The RSG Design Report (Reference 1) describes TRANFLOW as a code that solves the mass, energy, and momentum conservation equations for transient thermal/hydraulic phenomena using a fully implicit backward differencing technique. TRANFLOW was used for the steamline break accident, determining the pressure drops across SG secondary side internal components in a similar manner that was used for CEFLASH-4A as discussed in UFSAR Section 3.9.1.2.2.1.28.

TRANFLOW and CEFLASH-4B have been successfully used to evaluate the Westinghouse and Combustion Engineering (CE) fleet of steam generators, respectively. TRANFLOW is not a new code and has been used repeatedly by Westinghouse in steam generator replacements. Even though TRANFLOW has not been approved by the NRC for determining loads on SG internals during a steamline break, it has been shown to produce results comparable to NRC approved codes that were used for that purpose. Westinghouse LTR-NCE-05-145 (Reference 7) includes a comparison of TRANFLOW results to those of CEFLASH-4B and concludes that similar results were obtained by the two codes for secondary side depressurization of a steam generator. The following provides the basis to conclude that TRANFLOW results provide consistent and comparable results to CEFLASH-4A/CEFLASH 4B.

CEFLASH-4A is an NRC approved code for analyzing the reactor coolant system (RCS) blowdown transient performed by CE to provide thermal hydraulic data for use in Appendix K large break LOCA analyses. Analyzing RCS blowdown and decompression over the entire duration of an Appendix K large break LOCA analysis is not necessary to develop thermal hydraulic data for use in blowdown loads analyses. Only the hydrodynamics of the brief, initial phase of the transient are relevant.

For analyzing blowdown loads, CEFLASH-4A was modified to produce a code version designated as CEFLASH-4B. The modifications included removing CEFLASH-4A features that do not influence the course of the early decompression phase and adding nodes and volumes that expand the amount of spatial detail. CENPD-252-P-A (Reference 8) is the NRC approved topical report for CEFLASH-4B. It documents the results of a comparison of predicted CEFLASH-4A and CEFLASH-4B blowdown hydrodynamics during the brief, initial interval pertinent to developing data for blowdown loads analyses. CENPD-252-P-A states that the excellent agreement between the two codes' predictions confirms the similarity of CEFLASH-4A and CEFLASH-4B for blowdown loads analyses.

In LTR-NCE-05-145, a comparison of TRANFLOW and CEFLASH-4B was performed on the Watts Bar Unit 1 RSGs to show how these codes respond similarly for given break conditions. TRANFLOW like CEFLASH-4B applies a one-dimensional, two-phase, thermal-hydraulic analysis which calculates the time-dependent behavior of the fluid state resulting from a flow rupture. The RSGs were subjected to a postulated feedwater line break using TRANFLOW which were then compared to the CEFLASH-4B results. Results were obtained for a pertinent 0.25 second blowdown simulation initiated from steady-state conditions at 100% power. The same initial conditions and design parameters used in CEFLASH were used in TRANFLOW to make a valid comparison. The results obtained from the two codes during an FLB accident were compared and the differential pressure loads in the critical regions were observed. Peak pressure loads on steam generator internals agree favorably. Both codes showed a similar decrease in the divider plate pressure differential with increasing elevation and a reduction in the effect of the pressure wave across the SG support plates. The break flow rate also agreed quite favorably between the two codes. The comparison demonstrated that the two codes provide similar results during a secondary side depressurization of the steam generators.

Therefore, since CEFLASH-4A provides identical results to CEFLASH-4B, and TRANFLOW results are similar to CEFLASH-4B, then it can be concluded that TRANFLOW produces results comparable to those of CEFLASH-4A.

As described in Westinghouse LTR-NCE-04-28 (Reference 9), TRANFLOW is the workstation version of TRANFLO, a code originally developed for Westinghouse in the early 1970's to determine thermal-hydraulic conditions in the steam generators during various transients to assist in the design and structural analyses of steam generators. Westinghouse used TRANFLO to predict the pressure drops across the SG tube support plates in a utility's license amendment request to increase the lower limit of the voltage repair criteria for the steam generator tubes. During review, the NRC requested that the analyses be performed using RELAP 5 since TRANFLO had been approved for use in calculating mass and energy release to the containment following a steamline break, but not for detailed modeling of internal thermal-hydraulic conditions in a steam generator due to blowdown transients. The analyses were rerun using RELAP 5. Subsequently, Westinghouse performed comparisons of TRANFLO/TRANFLOW results to those of RELAP 5 which the NRC considered acceptable for calculating blowdown loads. Westinghouse LTR-NCE-04-28 concluded that based on detailed comparison of RELAP5 and TRANFLO/TRANFLOW, the two computer codes should be considered technically equivalent. TRANFLO/TRANFLOW results also agreed with the results of the NRC TRACE code.

As discussed in Westinghouse LTR-NCE-04-28 to demonstrate TRANFLO/TRANFLOW suitability for determining thermal-hydraulic conditions in the steam generators during various transients, TRANFLO/TRANFLOW has been developed and maintained under the Westinghouse Quality Assurance (QA) Program which is in compliance with the design control measures, including verification of 10 CFR 50, Appendix B. Under the Westinghouse QA program, TRANFLO/TRANFLOW is treated in the same manner as other computer programs used by Westinghouse in design and safety analyses, including programs that have been submitted to and approved by the NRC for use in safety analyses. The NRC has implicitly approved the use of TRANFLO/TRANFLOW, since it has been verified and validated consistent with the Westinghouse Quality Assurance program, and the NRC has reviewed and approved the elements of that program.

In addition to comparison to code predictions, TRANFLO/TRANFLOW verification has included comparison of TRANFLO results to test data and field data. TRANFLO results were generally in agreement with, or

conservative with respect to, test data.

Based on the code results comparison of TRANFLO/TRANFLOW to CEFLASH 4A and CEFLASH 4B for its use in predicting pressure drop during steamline breaks, the longstanding use of TRANFLO/TRANFLOW by Westinghouse on replacement SGs, and quality assurance program controls benchmarking performed, TRANFLO/TRANFLOW is concluded to be acceptable for its use on Waterford 3. TRANFLO/ TRANFLOW, as used on Waterford 3, meets the criteria of NEI 96-07, Section 4.3.8.1 that states that changes to elements of analysis methods that yield conservative results, or results that are essentially the same, would not be departures from approved methods.

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 #** 2012-04 / **Rev. #:** 0

Proposed Change / Document: Waterford-3 Cycle 19 Reload

Description of Change:

This evaluation addresses the Waterford 3 Cycle 19 reload that resulted from the Cycle 19 core design performed by Westinghouse [Reference 1]. The Cycle 19 PAD indentified several inputs and plant design data changes to the Cycle 19 reload analyses that had an adverse impact on the current Analysis of Record (AOR) results. Several Cycle 19 specific analyses had to be performed, incorporating these revised inputs, to determine whether the AOR remained bounding. Those analyses include:

- Inadvertent Boron Dilution event (results in a COLR change)
- Inadvertent Reactor Coolant Pump (RCP) Startup
- LBLOCA
- SBLOCA

In addition, to resolve a Cycle 19 Physics Checklist exception, a change to the ASI operating range for greater than 50% power is required in the COLR [Reference 2].

All Cycle 19 Reload specific results met the applicable acceptance criteria and remain bounded by the results of AOR.

Summary of Evaluation:

Inadvertent Boron Dilution Event
The Inadvertent Boron Dilution event previously required evaluation for Cycles 17 and 18 due to the placement of fresh and once-burned fuel on the core periphery. This affects the neutron source decay curve, which is burnup dependent, and impacts the Boron Dilution Alarm System (BDAS). The BDAS reset frequencies in the COLR were modified for Cycles 17 and 18 to maintain acceptable results for the analysis. Since the burnup of peripheral assemblies is greater than the limit specified in the AOR for Cycle 19 there is no need to maintain the Cycles 17 and 18 BDAS reset frequencies for Cycle 19. As such, the BDAS frequencies can revert to the less restrictive reset frequencies last presented in the Cycle 16 COLR, which remain applicable to the Boron Dilution AOR. Only the Inadvertent Boron Dilution event with the BDAS operable is impacted. There is no impact on the Inadvertent Boron Dilution event with the BDAS inoperable, since the neutron source decay curve is not used. A change to COLR 3.1.2.9 is made to reset the BDAS frequencies consistent with the Boron Dilution AOR.

Inadvertent Reactor Coolant Pump (RCP) Startup
The Physics Assessment Checklist showed that the low temperature most negative Isothermal Temperature Coefficient (ITC) for RCS temperatures less than or equal to 350°F did not meet the limits specified in the AOR. As a result, the Inadvertent RCP Startup event was evaluated for Cycle 19. The Inadvertent RCP Startup event is a reactivity balance event that does not use computer case modeling. The AOR considered a minimum stuck rod worth of 1.2% $\Delta\rho$ and an all rods in (ARI) shutdown margin of 1.0% $\Delta\rho$. Per the Cycle 19 Groundrules [Reference 3], parameter COR.45, the Cycle 19 minimum stuck rod worth is 1.3 % $\Delta\rho$. The expected minimum ARI shutdown margin for Cycle 19 is 1.5% $\Delta\rho$. Thus, accounting for only a portion of the increases in stuck rod worth and ARI shutdown margin, more than offsets the slight adverse impact on ITC for

[1] 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.

Cycle 19. The Inadvertent RCP Startup event remains bounded by the existing AOR with the evaluation outlined herein to account for the Cycle 19 exception on the ITC.

LBLOCA
The pin census used in the LBLOCA core wide oxidation calculation did not bound the Cycle 19 specific core pin census data over the full range of fuel rod peaking factors. As a result, the limiting core-wide oxidation case from the AOR was evaluated using the Cycle 19 specific core pin census data. The comparison of results showed that LBLOCA core-wide oxidation result from the AOR remained bounding compared to the Cycle 19 specific evaluation. The AOR core-wide oxidation remained bounding because the impact of higher oxidation produced by the Cycle 19 fuel pins that were above the AOR pin census for a small region of the core was more than offset by the lower oxidation produced by the Cycle 19 fuel pins that were below the AOR pin census for the remainder of the core.

The Fuel Performance data for the various fuel rod types in Cycle 19 changed. Also, the axial poison cutback region for Cycle 19 changed from 6 inches to 7 inches for some of the ZrB2 fuel rods. To assess the impact of these changes, the fuel rod design study from the LBLOCA AOR was rerun using revised Cycle 19 fuel performance data to ensure the worst case peak clad temperature was identified. The Cycle 19 specific analysis showed that same fuel type as in the AOR gave the worst case results. Therefore, the AOR was confirmed to be bounding and applicable to Cycle 19 operation with the revised fuel performance and ZrB2 IFBA fuel rod design with a 2x7 inch IFBA cutback.

SBLOCA
For Cycle 19 the plant design data values for the minimum HPSI pump flow rate versus RCS pressure changed relative to those utilized in the SBLOCA AOR for RSGs. A Cycle 19 specific analysis was run to assess the impact of the slightly higher (by less than 1 gpm over the entire flow curve) HPSI pump flow rate for Cycle 19. The net result was a reduction in the Peak Clad Temperature, Maximum Clad Oxidation, and Core Wide Oxidation for the limiting SBLOCA case of -27 °F, -1.5%, and -0.07%, respectively. The Cycle 19 SBLOCA analysis is bounded by the SBLOCA AOR for RSGs and continues to meet the ECCS acceptance criteria in 10CFR50.46.

Core Operating Limits Report (COLR)
As discussed above, a change to COLR 3.1.2.9 is being made to reset the BDAS frequencies consistent with the Boron Dilution AOR.

An ASI restriction is being placed on the COLSS Operable positive ASI limit of COLR 3.2.7. This restriction is required since the maximum radial power distribution of peripheral assemblies is greater than that used in the reactor vessel internals component heating analysis. The assessment using existing conservatisms was not able to completely offset the adverse impact of the Cycle 19 specific power distribution. Thus, an additional restriction on ASI was required for the results to remain bounded by the internals heating analysis. This results in a 0.03 reduction in the positive ASI limit when core burnup is greater than 250 EFPD.

References:

1. Letter, R. A. Loretz (Westinghouse) to C. Eastus (Entergy), "Waterford Unit 3 Cycle 19 Final Reload Analysis Report," NF-WTFD-12-17, August 20, 2012.

2. Letter, R. A. Loretz (Westinghouse) to C. Eastus (Entergy), "Startup Test and Setpoints Transmittal for Waterford-3 Cycle 19," NF-WTFD-12-32, November 30, 2012.

3. Waterford 3 Calculation ECS10-001, Rev. 2, "Waterford 3 Reload Analysis Groundrules", 12/14/2011.

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: Chris Eastus/ Chris Eastus / ESI / Fuels & Analysis / 12-06-2012
Name (print) / Signature / Company / Department / Date

Reviewer: Nasser Pazooki / Nasser Pazooki / ESI / Fuels & Analysis / 12-06-2012
Name (print) / Signature / Company / Department / Date

OSRC: Carl E. Rich Jr. / 12/7/12
Chairman's Name (print) / Signature / Date

W3 12-22
OSRC Meeting #

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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: The Cycle 19 Reload fuel does not initiate any accidents evaluated in the UFSAR. There are no changes to the Cycle 19 fuel design or fabrication that would have an impact on the frequency of occurrence of a fuel handling or mis-loading accident. None of the barriers to a fuel mis-loading are affected by the Cycle 19 Reload. The core and plant design changes are inputs to the analyses and will not increase the frequency of any event.

   Therefore, the Cycle 19 Reload will 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: Since fuel design for Cycle 19 core is essentially identical to the fuel design present in the Cycle 18 core, the Cycle 19 reload fuel will not require any equipment important to safety to be operated in a different manner or with a higher duty. No structure, system or component important to safety was changed as a result of Cycle 19 fuel design. Therefore, the probability of a malfunction of a structure, system or component important to safety is not impacted due to the introduction of the Cycle 19 core.

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

   BASIS: Cycle 19 reload safety analyses were evaluated/performed to assure that acceptance criteria are met for fuel performance, thermal-hydraulic performance, post-LOCA Emergency Core Cooling System (ECCS) performance, and non-LOCA event responses. Acceptable results from these analyses confirm that the core can be operated safely and meets all license requirements for accident response.

   There were several inputs and plant design data changes to the Cycle 19 reload analyses that had to be evaluated for possible adverse impact on the current Analysis of Record (AOR) results. Those analyses and the results are discussed in the above Summary of Evaluation section. Conservatisms in the stuck rod worth and ARI shutdown margin more than offset the slight adverse impact of the Cycle 19 ITC on the Inadvertent RCP Startup analysis. The LBLOCA core wide oxidation using the Cycle 19 core pin census resulted in a lower oxidation amount than the existing AOR. The LBLOCA results with Cycle 19 fuel performance data was shown to remain bounded by the LBOCA AOR. The SBLOCA analysis using the Cycle 19 revised HPSI flow rates showed a reduction in the peak clad temperature and oxidation relative to the AOR. All of the accident analyses were shown to be bounded by their existing licensing basis AORs and to meet the applicable acceptance criteria.

   The COLR changes ensure that W3 is operated in a manner that is consistent with the analysis assumptions. Boron Dilution Alarm Setpoint reset frequencies in COLR 3.1.2.9 are changed to be consistent with the Boron Dilution AOR. A more restrictive positive ASI after a core burnup of 250 EFPD is implemented in COLR 3.2.7 to maintain the assumptions of the reactor vessel internals heating analysis.

As a result of the COLR changes and the respective AORs remaining bounding for Cycle 19, there is no increase in the consequences of an accident evaluated in the UFSAR due to the Cycle 19 Reload.

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: Cycle 19 reload safety analyses were performed to assure that acceptance criteria are met for fuel performance, thermal-hydraulic performance, post-LOCA ECCS performance and non-LOCA transient response. These analyses confirm that the Cycle 19 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 19 analyses do not place greater reliance on any specific plant system, structure, or component to perform a safety function. No changes in the assumptions concerning equipment availability or failure modes have been made and none are necessary to implement Cycle 19. Thus, there is no increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR by the Cycle 19 reload.

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

BASIS: Cycle 19 reload does not introduce any new operating conditions, plant configurations or failure modes that could lead to a credible accident of a different type than any previously evaluated in the UFSAR. No accident initiator is impacted by Cycle 19 reload. Therefore, the possibility of an accident of a different type than any previously evaluated in the UFSAR will not be created.

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 Cycle 19 core design does not modify the design or operation of structures, systems, or components important to safety beyond the fuel itself. The reload core will not require any structure, system or component important to safety to be operated in a different manner or with a higher duty. Structures, systems and components important to safety will function in the same manner with the reload core as with the Cycle 18 reload core. The changes in core characteristics do not change any parameter that would affect the function of structures, systems or components important to safety. There are no new methods of failure associated with any of the changes associated with the Cycle 19 core.

Based on the above, the Cycle 19 core does not create the possibility of 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: Cycle 19 reload safety analyses were performed to assure that acceptance criteria are met for fuel performance, thermal-hydraulic performance, post-LOCA ECCS performance and non-LOCA response. These analyses confirm that the core can be operated safely and can be expected to meet license requirements for accident response. The Cycle 19 reload safety analyses were performed to demonstrate compliance with the existing design basis limits for the fuel cladding, RCS pressure boundary and containment fission product barriers.

All events have been evaluated in the reload analysis to assure that they meet their respective criterion for Cycle 19. Based on a review of the reload analysis results, the design basis and regulatory limits for the fuel cladding, RCS pressure boundary, and containment will not be exceeded for Cycle 19.

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: There were no changes to any of the methodologies, described in the UFSAR, used in establishing the design bases or in the safety analyses for Cycle 19 reload. There is no requirement for any Technical Specifications (TS) changes as a result of Cycle 19 reload. All the analyses were performed using NRC approved methods. Therefore, there is no departure from a method of evaluation described in the UFSAR used in the safety analyses for Cycle 19.

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: WF3 **Evaluation # 2012-05 / Rev. #:** 0

Proposed Change / Document: EC-41095

Description of Change:

The Dry Cooling Towers (DCTs) and the Wet Cooling Towers (WCTs) are located outside the Reactor Auxiliary Building on the east and west sides. The DCTs are the primary heat sink for the CCW system during normal operation. During a LOCA (accident mode), both dry towers will operate in conjunction with the CCW heat exchangers to remove all heat required for a safe shutdown. By design, the DCTs remove approximately 60% of the total heat load rejected by the ultimate heat sink during an accident. The WCTs will operate to assist CCW in heat removal until the cooling load is reduced to a level that the dry towers can handle alone. The Ultimate Heat Sink (UHS) consists of dry and wet cooling towers and water stored in the wet cooling tower basins.

The Essential Chillers provide chilled water (CHW) for space cooling purposes and rejects heat through the Component Cooling Water (CCW) system to the UHS. Auxiliary Component Cooling Water (ACCW) system can be used for CHW heat rejection if CCW loses heat rejection capacity. The Essential Chiller Supply and Return Isolation Valves, CC-301A(B), CC-322A(B), ACC-112A(B), and ACC-139A(B), are located on the +21ft elevation of the Reactor Auxiliary Building in the overhead of the CCW Heat Exchanger A and B rooms. Cooling water flow is required to be supplied to the condenser sections of the running essential chillers during both normal operating and accident conditions. CCW is normally aligned to provide this flow. ACCW is automatically aligned in place of CCW during conditions of high heat load on the CCW system.

As an alternative to local manual hand-wheel operation, high pressure air bottles will be staged near Nitrogen Accumulators Number 1 and 2, which may be used to re-pressurize the accumulators when required for stroking the Essential Chiller Supply and Return Isolation Valves, CC-301A(B), CC-322A(B), ACC-112A(B), and ACC-139A(B). The Nitrogen Accumulators are sized to provide control air to these valves(including ACC-126A(B)) for ten (10) hours following a loss of Instrument Air. Since all the valves operated from these accumulators are of the fail-as-is type, the accumulator could be allowed to expire and the system left to operate until such time as the DCT would take over all loads. Manual flow reduction may be required to preserve water inventory. Manual action would then re-align the system to the DCT. At that point, the ACCW pump is secured and cooldown would continue using the DCT and the CCW system which would be manually aligned (4 valves to operate) or the nitrogen bottles will be recharged and the valves will be operated from control room. In the worst case accident, radiation levels would be at less than 2 R/hr, however, stay time would be very short for operating the valves or charging the accumulators (an engineering estimate is 15 minutes) and dose is considered acceptable. (reference: W3-DBD-014)

ACC-126A and ACC-126B function to maintain CCW temperature below 95°F during normal operation and 115°F for a mission time of 30 days post accident. These valves and accumulators are accessible for manual operation or recharging of accumulators, respectively, if required to preserve water inventory.

ACC-112A, ACC-112B, ACC-139A, ACC-139B, CC-301A, CC-301B, CC-322A and CC-322B function to maintain valve alignment for cooling water supply to the essential chillers from CCW whenever the CCW temperature is 102°F or less, and to change the alignment to supply cooling water from ACCW whenever CCW temperature exceeds 102°F. These valves have a mission time of 7 days post accident.

A very reliable design will be employed consisting of two compressed air bottles for each nitrogen accumulator sized to fill the accumulators without over pressurizing and without the need for a regulator or a relief valve. The tubing and air bottles will be permanently mounted with two isolation valves and a vent valve for an operator to re-position to recharge each nitrogen accumulator.

[1] 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.

Operations will provide procedures for the use of the air bottles to charge the accumulators when needed.

NRC INFORMATION NOTICE 97-78 [Reference 2] lists the following requirements for crediting an operator action.
The original design of nuclear power plant safety systems and their ability to respond to design-basis accidents were described in licensees' FSARs and were reviewed and approved by the NRC. Most safety systems were designed to rely on automatic system actuation to ensure that the safety systems were capable of carrying out their intended functions. In a few cases, limited operator actions, when appropriately justified, were approved. Proposed changes that substitute manual action for automatic system actuation or modify existing operator actions, including operator response times, previously reviewed and approved during the original licensing review of the plant will, in all likelihood, raise the possibility of a USQ. Such changes must be evaluated under the criteria of 10 CFR 50.59 to determine whether a USQ is involved and whether NRC review and approval is required before implementation. A licensee may not make such changes before it receives approval from the NRC when the change, test, or experiment may (1) increase the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously analyzed in the FSAR, (2) create the possibility of an accident or a malfunction of a different type than any previously evaluated in the FSAR, or (3) reduce the margin of safety as defined in the basis for any TS. In the NRC staff's experience, many of the changes of the type described above proposed by licensees do involve a USQ.

NRC INFORMATION NOTICE 97-78 also lists specific requirements the NRC will use to review new operator actions. Based on these guidelines, the NRC's reviews of licensees' analyses typically include, but are not limited to, (1) the specific operator actions required; (2) the potentially harsh or inhospitable environmental conditions expected; (3) a general discussion of the ingress/egress paths taken by the operators to accomplish functions; (4) the procedural guidance for required actions; (5) the specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions; (6) any additional support personnel and/or equipment required by the operator to carry out actions; (7) a description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken; (8) the ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery; and (9) consideration of the risk significance of the proposed operator actions. These requirements are addressed in the response to Question 3 below.

Summary of Evaluation: The installation of backup air stations to re-pressurize Nitrogen Accumulators 1 & 2 is determined to not require prior approval by the NRC. A description of this plant system will be added to UFSAR section 9.3.9.2.1.

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: William Day/ See AS for signature and date
Name (print) / Signature / Company / Department / Date

Reviewer: Nasser Pazooki / See AS for signature and date
Name (print) / Signature / Company / Department / Date

OSRC: Carl E. Richter Jr 12/7/12
Chairman's Name (print) / Signature / Date

W3 12-23
OSRC Meeting #

References:
CR-WF3-2012-06703
B430 sh V32
B424 sh 827
B424 sh 828
1564-4045
TD-B237.0155_EC-23632 markup
ECM89-002, Nitrogen Accumulator Leakage Criteria
W3-DBD-014, Safety Related, Air Operated Valves
SQ-MN-245, Nitrogen Accumulator

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: The safety function of the Ultimate Heat Sink, which consists of the Dry Cooling Towers (CCW) and Wet Cooling Towers (ACCW) and the water stored in the Wet Cooling Tower basins, is to mitigate the consequences of an accident by dissipating the heat removed from the reactor and its auxiliaries after a design basis accident. The CCW and ACCW systems are not considered initiators for any accident described in the UFSAR.

   The compressed air bottles installed by this modification are totally independent of the Instrument Air system. Nitrogen Accumulators 1 and 2 supply backup motive gas for safety train A and B valves respectively to support the Essential Chillers. There is an independent seismically qualified compressed air bottle station being installed for each Nitrogen Accumulator to maintain train separation. The compressed air bottles are separated from their respective nitrogen accumulator by two isolation valves with an open vent valve between them to prevent leakage from the compressed air bottles from over pressurizing the nitrogen accumulators. Charging the nitrogen accumulators will be administratively controlled to ensure that the compressed air bottles are not aligned with the nitrogen accumulators unless the nitrogen accumulator pressure is less than or equal to 80 psig. When a nitrogen accumulator is charged with a starting pressure of 80 psig, the maximum calculated pressure is 440.1 psig after the compressed air bottles have fully discharged. The design pressure of the nitrogen accumulators is 800 psig which provides a large margin to prevent over pressurizing a nitrogen accumulator. Additionally, it has been shown by calculation in SQ-MN-245 that Nitrogen Accumulator 1 & 2 are qualified for a design pressure of 2525 psig. This combination of backup compressed air station design with administrative controls ensures that the nitrogen accumulators are not over pressurized.

   Since the ACCW system is not an accident initiator and the possibility of over pressurizing a nitrogen accumulator is prevented, this change does not affect 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 safety function of the Ultimate Heat Sink, which consists of the Dry Cooling Towers (CCW) and Wet Cooling Towers (ACCW) and the water stored in the Wet Cooling Tower basins, is to mitigate the consequences of an accident by dissipating the heat removed from the reactor and its auxiliaries after a design basis accident. The proposed change provides an alternate method of manual action to operate the ACCW and CCW valves. As discussed earlier the alternate method is to recharge the accumulators that would allow the valves to be remotely operated from the control room rather than manually operating the valves using the handwheel. Specifically, this change will require manually opening two small isolation valves and closing one small vent valve to recharge each accumulator as opposed to manually turning the handwheels on four separate large valves to swap Essential Chiller cooling from ACCW to CCW. Operators would then shift Essential Chiller cooling water from ACCW to CCW using a control switch in the Control Room. This method results in less operator involvement in valve manipulation and therefore is considered more reliable in lieu of operator manual manipulation of the valves.

   The compressed air bottles installed by this modification are totally independent of the Instrument Air system. Nitrogen Accumulators 1 and 2 supply backup motive gas for safety train A and B valves respectively to support the Essential Chillers. There is an independent seismically qualified compressed air bottle station being installed for each Nitrogen Accumulator to maintain train separation. The compressed air bottles are separated from their respective nitrogen accumulator by two isolation valves with an open vent valve between them to prevent leakage from the compressed air bottles from over

pressurizing the nitrogen accumulators. Charging the nitrogen accumulators will be administratively controlled to ensure that the compressed air bottles are not aligned with the nitrogen accumulators unless the nitrogen accumulator pressure is less than or equal to 80 psig. When a nitrogen accumulator is charged with a starting pressure of 80 psig, the maximum calculated pressure is 440.1 psig after the compressed air bottles have fully discharged. The design pressure of the nitrogen accumulators is 800 psig which provides a large margin to prevent over pressurizing a nitrogen accumulator. Additionally, it has been shown by calculation in SQ-MN-245 that Nitrogen Accumulator 1 & 2 are qualified for a design pressure of 2525 psig. This combination of backup compressed air station design with administrative controls ensures that the nitrogen accumulators are not over pressurized.

Based on above, the proposed method does not increase the likelihood of occurrence of the malfunction of the Ultimate Heat Sink or the nitrogen accumulators as 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: The safety function of the Ultimate Heat Sink, which consists of the Dry Cooling Towers (CCW) and Wet Cooling Towers (ACCW) and the water stored in the Wet Cooling Tower basins, is to mitigate the consequences of an accident by dissipating the heat removed from the reactor and its auxiliaries after a design basis accident. This change allows for a more reliable operation of the valves, when required, to ensure that the Essential Chillers are realigned to the CCW system at the appropriate time. If the re-alignment to CCW is not performed there is a potential increase in the consequences of an accident. As discussed in question 2, the proposed change to recharge the valve accumulators to ensure the valves will be able to stroke when required to support swapping the Essential Chillers from the ACCW system to the CCW system to support continued heat removal from the Chillers is more reliable with a backup air system that allows control room valve manipulation. When the accumulators are recharged the valves will be able to stroke as they normally do. This is required up to eight days after an accident. To ensure the availability and ease of recharging the accumulators, compressed air bottles are stationed near the accumulators for operators to use to recharge the accumulators when needed. Additionally, operators will be trained on performing this task and will be briefed on the need to do this in order to operate the valves more than 10 hours after an accident (note that swapping of chillers from WCT to DCT does not need to occur until 7 days into the event). Therefore, the proposed change does not result in an increase in the consequences of an accident previously evaluated in the UFSAR.

The alternate method of manual operator action is evaluated against NRC Information Notice 97-78 and ANSI/ANS-58.8-1994 on system operation post RAS. The ten primary attribute evaluations are specifically listed below.

(1) The specific operator actions required;
The operator actions required to recharge the nitrogen accumulator number I and II using permanently installed compressed air bottles will be added to procedure OP-003-016, Instrument Air. The operator will be required to open two isolation valves and close one vent valve to align the permanently mounted compressed air bottles with their associated nitrogen accumulator. This is a simple task for operators to perform. No adverse impact.

(2) The potentially harsh or inhospitable environmental conditions expected;
The nitrogen accumulators number 1 and 2 are located on the +21 of the RAB inside the CCW Heat Exchanger rooms. In the worst case accident, radiation levels would be less than 2 R/hr, however, stay time would be very short for operating the valves or charging the accumulators (an engineering estimate is 15 minutes) and therefore, the dose to the operator is considered acceptable and less than manual CCW/ACCW valve actuator positioning in overhead areas. No adverse impact.

(3) A general discussion of the ingress/egress paths taken by the operators to accomplish functions;
The action taken outside the control room will be in the RAB +21 elevation inside the CCW Heat Exchanger Rooms. Emergency lighting is available in all locations needed and the worst case ambient temperature in the CCW Heat Exchanger Rooms is 107 °F, so the environment conditions

will not be adverse. Operators currently have to enter these areas post accident to manually operate ACC-126A(B). This is a frequently traveled path with no significant obstacles. No adverse impact.

(4) The procedural guidance for required actions;
Procedural guidance for charging nitrogen accumulator number I and II using permanently installed compressed air bottles will be added to procedures OP-901-511, Instrument Air Malfunction, and OP-003-016, Instrument Air.

(5) The specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions;
Operators will be trained on the alternate manual actions to recharge Nitrogen Accumulators 1 & 2. TEAR W3-2012-1267 and Simulator DR 12-0157 will evaluate the impact to training programs and simulator. The new valves to be manipulated are located in accessible areas from the RAB +21 elevation floor. (no ladders or climbing necessary)

(6) Any additional support personnel and/or equipment required by the operator to carry out actions;
The operations personnel and equipment that address the current operation will not change. No additional support personnel or equipment are required to carry out the actions to charge the nitrogen accumulators. The air bottles, tubing, fittings and valves to support recharging the nitrogen accumulators are permanently mounted. Operators only have to open two isolation valves and close one vent valve to charge each nitrogen accumulator. No adverse impact.

(7) A description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken;
This operator action will not be required until well into the event (approximately 7 days). At this point in the event the TSC will be monitoring the Ultimate Heat Sink, CCW temperature and water inventory and will be able to alert operators when the action is required. There is instrumentation in the control room that provides CCW temperature indication that operators will monitor to determine when the action is required. There is also instrumentation in the control room that monitors flow in the ACC system. In addition, local pressure indication for the compressed air bottles and the nitrogen accumulators provide operators with the information needed to determine if the air bottles have discharged and pressurized the nitrogen accumulator. No adverse impact.

(8) The ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery;
The alternate method i.e., recharging the accumulators is relatively simple, straightforward and requires very few (3) manipulations by operators (open two isolation valves and close one vent valve for each accumulator). Should an operator begin to open an isolation valve prior to closing the vent valve, it will be obvious as gas will issue from the vent valve and the operator can immediately close the isolation valve. The nitrogen accumulator pressure indicator is seismically qualified and considered reliable. In order to over pressurize a nitrogen accumulator, an operator would have to not observe that nitrogen pressure is greater than 80 psig in the accumulator and manipulate three valves in error. Also, by design, as shown in calculation ECM89-002, the nitrogen accumulators cannot be pressurized beyond allowable design pressures even if the backup accumulators are discharged into a nitrogen accumulator containing 800 psig nitrogen pressure. Recovery from errors is expected to be very short with respect to the time available to perform the action. No adverse impact.

(9) Consideration of the risk significance of the proposed operator actions;
The proposed actions to charge the nitrogen accumulators, open two isolation valves and close one vent valve, is considered a simple task with minimal risk. The action to recharge a nitrogen accumulator is safer than local operation of the valves manual handwheels as no climbing is required. No adverse impact.

(10) Time response as outlined in ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-

Related Operator Action";
ANSI/ANS-58.8-1994 provides time requirements for different accident scenarios. In general it allows operator actions that are not required until after the initial 30 minutes following a design basis event to be performed outside of the control room. Since this action is expected to be required 7 days after the event, there is adequate time to identify, diagnose and perform the action. These actions are allowed, provided 1) the environment is safe for the operator 2) the operator has adequate communication links with the Control Room 3) there is adequate emergency lighting 4) the operator has procedures for performing a safety related task and 5) the operator has the tools and equipment necessary to perform the safety related actions.

1) The ambient temperatures in the RAB +21 CCW Heat Exchanger rooms are expected to be acceptable for operations activities ($\leq$107 °F). The dose rates will be less than 2 R/hr which is acceptable for short duration tasks. (ref: DBD-014, G-M-005)
2) Operators have radio communication to contact the control room. Additionally phones are located near the CCW heat exchanger rooms.
3) Emergency lighting is provided in the CCW Heat Exchanger rooms and the RAB +21 path normally traveled to get there. Temporary portable lights (flashlights) could be used as backup.
4) There will be specific procedural guidance to direct the operators how to charge the nitrogen accumulators using the Backup Air Supply (bottle) system.
5) The compressed air bottles along with the necessary fittings, connections and isolation valves will be permanently mounted. The operators will have to open two isolation valves and close one vent valve to recharge each accumulator.

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 proposed activity allows for simple manual recharge of the accumulators that would allow post-accident remote operation of the Essential Chiller cooling water supply and return valves for an extended time frame if required. This action has no impact on the valves or the way they operate. Therefore, failure modes for the valves and system as discussed in the UFSAR are not impacted by this change. This change only impacts the ability to remotely operate these valves several hours to days after a LOCA. This modification preserves the ability to manually operate the Essential Chiller Supply and Return cooling water valves eight days into an accident as described in FSAR 9.2-2, thus assuring safe shutdown and protection of fission product barriers. Therefore, the proposed change has no impact on consequences of malfunction of these valves as evaluated in the UFSAR, therefore does not impact the consequences of a malfunction of an SSC 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 safety function of the Ultimate Heat Sink, which includes the ACCW system, is to mitigate the consequences of an accident by dissipating the heat removed from the reactor and its auxiliaries after a design basis accident. The ACCW system is not considered an initiator for any accident described in the UFSAR. The backup compressed air stations will not be used unless there has already been a design basis event and then not until sometime after the event has occurred. They are passive standby systems with no credible method of initiating an accident. Therefore, this change does not create the possibility of 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 proposed activity allows for simple manual recharge of the accumulators that would allow the post-accident remote operation of the Essential Chiller supply and return cooling water valves for longer time if required. This action has no impact on the valves or the way they operate. The failure modes for the valves and system as discussed in the UFSAR are not impacted by this change. This change only enhances the ability of the valves to perform their post-accident safety function for much longer if needed The proposed change has no impact on any failure modes or malfunction of these valves as discussed in the UFSAR.

As discussed earlier, the design of the backup air stations along with administrative controls ensures that the nitrogen accumulators are not over pressurized .

Therefore it does not create a possibility for a malfunction of an SSC with a different result than 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 proposed activity enhances the ability of the UHS system to perform its post-accident safety function. Therefore, the proposed change does not adversely impact the performance of the UHS system that could impact the performance of the fission product barrier such as containment. Therefore, no fission product barriers are adversely affected by this change. This change implements an alternate method to manually operating valves associated with the essential chillers cooling water. In lieu of manually operating valves, the nitrogen accumulators 1 and 2 will be re-pressurized so that the normal remotely operated valve controls may be used to position the air operated valves associated with these accumulators ten or more hours (typically 7 days) following a design basis accident. This results in the ACCW system and the Essential Chillers operating as they normally would if there were no loss of instrument air. Therefore, the proposed change does not result in a design basis limit for a fission product barrier as described in the UFSAR to 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: This change involves the Ultimate Heat Sink and the ACCW system. Specifically it impacts the supply water for the Essential Chillers following an accident and the ability to switch from ACCW to CCW as well as the CCW heat exchanger outlet temperature control valves. There is no change in the method of operation of the UHS or its assumed performance in the accident analyses. No new analysis is being performed for the UHS as a result of 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 # 2012-06 / Rev. #: 0**

Proposed Change / Document: EC-40444

Description of Change: Currently Waterford 3 post accident dose analyses do not account for any steam leakage, i.e., no steam leakage, from main steam safety valves (MSSV) and atmospheric dump valves (ADV). Past experience has shown that MSSVs/ADVs could have minor leakages that should be included in the accident dose analyses. A search of past condition reports indicated that the largest MSSV/ADV leakage from both steam lines, estimated to be 92.7 lb/hr, occurred during Cycle 18 operations. This leakage was abnormally high due to MSIV "A" (MS-124A) packing leakage and ADV#2 (MS-116B) seat leakage which accounted for about 83.8% of the total leakage. During Refuel 18 maintenance work was performed on both ADV#2 and MSIV "A" valves to address excess leakage.

To account for MSSV/ADV valve leakages in accident dose analyses, the proposed activity incorporates a conservative combined MSSV/ADV steam leakage of 280 lb/hr per steam line into the Waterford design basis post-accident radiological dose analyses. The assumed combined MSSV/ADV leakage of 280 lb/hr per steam line is well above the total MSSV/ADV leakages per steam line that has been experienced in the past. The 280 lb/hr leakage can be applied to any or combination of MSSVs/ADVs on each steam line.

The analysis of the radiological dose consequences of the events impacted by MSSV/ADV leakage assumes loss of offsite power and in turn credits cooldown of the plant via steaming from ADVs to shutdown cooling (SDC) entry conditions. Note that for a large break LOCA due to releases to the containment building, depressurization of the RCS, and RCS cooldown via safety injection system, releases to the secondary side are considered negligible. Thus, the impact of ADV/MSSV leakage would be negligible for large break LOCA. Previous Waterford 3 dose analyses assume all steam releases to the environment stop after SDC entry conditions are achieved. However, if MSSV/ADV leakage is assumed, the leakage will continue for several hours past SDC entry, until cold shutdown conditions are achieved. Therefore the following events are re-analyzed with a combined MSSV/ADV steam leakage of 280 lb/hr per each steam line up to cold shutdown conditions.

- Small Break LOCA
- Main Steam Line Break Inside Containment
- Steam Generator Tube Rupture
- Outside Containment Main Steam Line Break / Feedwater Line Break
- Control Element Assembly Ejection
- Reactor Coolant Pump Seized Rotor / Sheared Shaft
- Inadvertent Atmospheric Dump Valve Opening
- Excess Main Steam Flow with Loss of Offsite Power

[1] 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.

- Letdown Line Break

The constant 280 lb/hr leakage from each steam line is a conservative assumption for dose calculation purpose. This leakage is applied during normal operating conditions. As the plant cools down to SDC entry and from SDC entry to cold shutdown the leakage from MSSV/ADV will continue to decrease accordingly. The MSSV/ADV leakage from event initiation to SDC entry will help plant cooldown to SDC entry and therefore will not result in any additional emergency feedwater (EFW) requirement. The steam leakage beyond SDC will also continue to contribute to plant cooldown and as the plant continues to cooldown the leakage continues to decrease. Therefore, the impact of the leakage on overall water consumption is considered to be negligible.

Summary of Evaluation:
The evaluation demonstrates that incorporating a combined MSSV/ADV steam leakage of 280 lb/hr per each steam line into Waterford 3 design basis post-accident radiological dose calculations does not result in any of the regulatory dose limits to be exceeded, does not result in any of the dose results documented in the FSAR to be exceeded and does not result in an un-reviewed safety question, i.e., does not require prior NRC approval. The impact of leakage on water consumption is negligible. Therefore, including a combined MSSV/ADV steam leakage of 280 lb/hr per each steam line into post-accident dose analyses is acceptable.

References:
1. 1564.105A, Main Steam Safety Valve (MSSVs) Design Specification
2. 5817-828, Atmospheric Dump Valve Data Sheet
3. Waterford 3 Calculation ECS04-013, Small Break Loss of Coolant Accident (SBLOCA) Alternative Source Term (AST) Radiological Dose Consequences for 3716 MWt Extended Power Uprate (EPU).
4. Waterford 3 Calculation ECS04-004, Main Steam Line Break (MSLB) Inside Containment Alternative Source Term (AST) Radiological Dose Consequences for 3716 MWt Extended Power Uprate (EPU).
5. Waterford 3 Calculation ECS04-002, Steam Generator Tube Rupture (SGTR) Alternative Source Term (AST) Radiological Dose Consequences for 3716 MWt Extended Power Uprate (EPU).
6. Waterford 3 Calculation ECS04-006, Feedwater Line Break (FWLB) Alternative Source Term (AST) Radiological Dose Consequences for 3716 MWt Extended Power Uprate (EPU).
7. Waterford 3 Calculation ECS04-003, CEA Ejection Alternative Source Term (AST) Radiological Dose Consequences for 3716 MWt Extended Power Uprate (EPU).
8. Waterford 3 Calculation ECS04-008, Letdown Line Break Alternative Source Term (AST) Radiological Dose Consequences for 3716 MWt Extended Power Uprate (EPU).
9. Westinghouse Letter LTR-TDA-12-28, Evaluation of Waterford-3 for Increased Radiological Releases Due to Main Steam Valve Leakage, Dated 7/19/2012.

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: Nasser Pazooki / [signature] / ESI / Fuels & Analysis / 12/20/12
Name (print) / Signature / Company / Department / Date

Reviewer: D. W. Fouts / [signature] / ESI / Fuels & Analysis / 12-20-12
Name (print) / Signature / Company / Department / Date

OSRC: Keith Nichols / [signature] 12/29/12
Chairman's Name (print) / Signature / Date

OSRC Meeting # W3 12-26

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:

The proposed activity incorporates a combined MSSV/ADV steam leakage of 280 lb/hr per each steam line into the Waterford design basis post-accident radiological dose analyses. This leakage assumption in the dose analyses does not involve a change to any of the facility equipment or the manner in which they are operated and has no impact on any accident initiator. Therefore, the proposed change does not result in an 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 proposed activity incorporates the assumption of a combined MSSV/ADV steam leakage of 280 lb/hr per each steam line into the Waterford design basis post-accident radiological dose analyses. The proposed change is not involved in any physical changes to the MSSVs and ADVs, or the manner in which they are operated. Therefore the proposed change has no impact on the likelihood of occurrence of a malfunction of MSSVs/ADVs or any other SSC important to safety as 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:

As discussed in response to previous questions, an assumption of a combined MSSV/ADV steam leakage of 280 lb/hr per each steam line is included in the impacted post-accident radiological dose analyses. NEI 96-07, "Guidelines for 10CFR50.59 Implementation", defines minimal increase as an increase to the existing calculated offsite or control room dose results which is less than or equal to 10% of the available dose margins (difference between existing dose results and the regulatory acceptance value).

The increase in the radiological dose consequences for all the events re-analyzed with an assumed 280 lb/hr MSSV/ADV steam leakage per steam line were evaluated against available existing dose margins. The evaluation demonstrated that the increase in offsite and control room dose consequences were less than 10% of the available margins in the existing dose results. 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:

The proposed change does not involve any physical change to the MSSVs or ADVs or the manner in which they operate. The consequences of malfunction of these valves are not impacted by the proposed change. Therefore, there is no increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR, as a result of the proposed change.

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

BASIS:

The proposed change does not involve any physical change to the MSSVs or ADVs or the manner in which they operate. Therefore, the proposed change does not 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 proposed activity incorporates an assumed combined MSSV/ADV steam leakage of 280 lb/hr per each steam line into the Waterford design basis post-accident radiological dose analyses. The proposed change does not involve any physical change to the MSSVs or ADVs or the manner in which they operate and therefore does not create the possibility of a different malfunction than previously evaluated. Therefore, the proposed change does not create a possibility for a malfunction of an SSC 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 proposed change incorporates an assumed combined MSSV/ADV steam leakage of 280 lb/hr per each steam line in the post-accident radiological dose analyses. This change does not involve any physical change to the MSSVs or ADVs or the manner in which they perform their post-accident safety function. This activity also does not involve any physical change to any fission product barriers or the way they operate.

Therefore, the proposed change has no impact on post accident performance of fission product barriers and does not result in the design basis limits for these barriers to 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 proposed change resulted in the re-analyses of several radiological dose analyses for several events described in the FSAR. All the analyses were performed using the current approved methodology. Therefore, the proposed change did 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 # 2012-07/ Rev. #** 0

Proposed Change / Document: EC 25199

Description of Change:

The subject changes revise the Waterford 3 Safety Injection Sump Strainer design basis as a result of Steam Generator Replacement and Nuclear Regulatory Commission (NRC) Generic Letter 2004-02 "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors" along with associated NRC Request for Additional Information (RAI) responses. The replacement Steam Generators are supplied with insulation termed non problematic (Reflective Metal Insulation) as opposed to fibrous Nukon and Metal Encapsulated Insulation that cover the original Steam Generators (Reference EC8437). NRC RAI correspondence credited new Safety Injection Sump strainer head loss testing which utilized reduced debris loading and a more conservative testing methodology. Engineering report WF3-ME-10-00006 (EC 26496) documents the new testing performance results for the design basis sump strainer head loss. Methodology changes to account for water hold-up in the Reactor Coolant System requires revision to MNQ6-4 "Water levels in Containment", and are also based on the GL 2004-02 RAIs.

This 50.59 Evaluation will address the potentially adverse impacts due to the sump testing documented in calculation WF3-ME-10-00006, which reduces the Net Positive Suction Head (NPSH) available for the High Pressure Safety Injection (HPSI), Low Pressure Safety Injection Pumps (LPSI) and Containment Spray (CS) system pumps while in the Recirculation Actuation Signal (RAS) mode of operation taking suction off the Safety Injection sump.

In order to respond to the NRC RAIs, new head loss testing utilizing the reduced debris loads and a more conservative testing method was deemed necessary and performed. The testing results for the new head loss are documented in engineering report WF3-ME-10-00006. The resultant increases in sump strainer head loss requires reflection in NPSH calculations ECM07-001 and ECM91-011 which are revised under this EC. Containment minimum water level calculation MNQ6-4 is also revised to include accounting for water hold-up in the Reactor Coolant System as required supporting the RAI discussions / resolutions. The net impact to the Waterford 3 Safety Injection Sump Strainer design basis is that strainer head loss increases which reduces the HPSI, LPSI and CS pumps' NPSH available margin while taking suction from the Safety Injection Sump.

The Safety Injection System is designed to inject borated water into the Reactor Coolant System (RCS) to flood and cool the reactor core and to provide for heat removal from the reactor core for extended periods following a Loss of Coolant Accident (LOCA). The system also injects borated water into the RCS to increase the shutdown margin following a rapid cool down of the RCS due to a Main Steam Line Break (MSLB). The HPSI and LPSI pumps start automatically upon receipt of a Safety Injection Actuation Signal (SIAS). Initially, pump suction is supplied by the RWSP. Once the RWSP reaches 10% indicated level, a RAS is generated during which the injection mode of operation ends and the recirculation mode begins. In the recirculation mode, suction is taken from the Safety Injection Sump. At this time the HPSI pump suction is diverted to the Safety Injection Sump and the LPSI pumps are secured.

The Containment Spray System is designed to remove heat and fission products from the containment atmosphere during and following either a LOCA or MSLB inside the containment. The system also limits off-site radiation by reducing the pressure differential between containment and the external environment and scrubbing the containment atmosphere following a LOCA so that the offsite dose and the dose to the

[1] 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.

Operators in the Control Room are within regulatory limits. Containment Spray is initiated following a major LOCA or MSLB in the containment by a Containment Spray Actuation Signal (CSAS) which results from the combination of a high-high containment pressure signal and a SIAS. Initially, CS pump suction is supplied by the RWSP. Once the RWSP reaches 10% indicated level, a RAS is generated during which the injection mode of operation ends and the recirculation mode begins. In the recirculation mode, suction is taken from the Safety Injection Sump.

The proposed change only affects the recirculation mode of operation during which debris generated due to a LOCA will transport to and collect on the Safety Injection Sump Strainers causing head loss across the strainers to increase and result in a decrease of NPSH available for the operating pumps. Regulatory Guide 1.1 requires that sufficient NPSH be provided to prevent pump cavitation for the injection and recirculation modes. Regulatory Guide 1.82 provides guidelines for evaluating the adequacy of the availability of the sump and suppression pool for long-term recirculation cooling following a LOCA.

Design and analysis of the Waterford 3 Safety Injection Sump is in accordance with Regulatory Guide 1.1 and 1.82 with a few exceptions currently discussed in the UFSAR. The exceptions are required as Regulatory Guide 1.82 was issued after the Waterford 3 Sump was initially designed. The proposed change does not affect any of the previously NRC approved exceptions.

Regulatory Guide 1.82 states that in order for a centrifugal pump to perform its safety function, there must be adequate margin between the available NPSH and the required NPSH. Failure to provide and maintain adequate NPSH margin for the containment heat removal system / pumps could result in pressurization of the containment building above design pressure, potential failure with an increase in offsite and control room radiological dose.

Summary of Evaluation:

The increased head loss evaluated by this 50.59 results from revised strainer debris loading resulting from testing (ref. Engineering Report WF3-ME-10-00006). Debris loads have been determined utilizing the NRC approved method discussed in NEI 04-07 Volumes 1 and 2. Testing was performed using a protocol in accordance with NRC issued recommendations. As a result of the new testing, the Safety Injection Sump Strainer head loss has increased from the previous value of 0.377 ft-water at 210 °F to 2.155 ft-water at 210 °F. This increase in head loss results in a limiting minimum NPSH margin of 0.329ft for HPSI pump B, 9.203ft for LPSI Pump A and 4.417ft for CS pump B. HPSI A, AB, LPSI B and CS A pumps have greater NPSH margins than the limiting values. UFSAR section 6.2.2.3.2.1 currently list the limiting NPSH margin for the CS pumps as 6.19 ft-water and section 6.3.2.2.2.3 list the limiting NPSH margin for the HPSI pumps as 2.34 ft-water. Both UFSAR section are being updated to reflect the revised NPSH values. As the limiting NPSH margins for both the HPSI and CS pumps are greater than zero, there continues to be assurance that pump performance will remain acceptable to ensure adequate core cooling.

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 ☐ Yes ☒ No *require prior NRC approval?*

Preparer: Thomas R. Hempel / Thomas R. Hempel / EOI / Design Engineering / 12-19-12
Name (print) / Signature / Company / Department / Date

Reviewer: Greg Ferguson / Greg Ferguson / EOI / Design Engineering / 12-29-12
Name (print) / Signature / Company / Department / Date

OSRC: K. Nichols / K. Nichols 12/29/12
Chairman's Name (print) / Signature / Date

OSRC Meeting # W3 12-26

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:
   The proposed change evaluates the adverse impact of sump testing documented in Engineering Report WF3-ME-10-00006 on SI and CS pump NPSH available margins. This change does not involve any physical changes to these pumps or the manner in which they are operated. The evaluation demonstrates that the available NPSH margins are sufficient for the pumps to perform their post accident safety function. Therefore, the proposed change has no impact on any accident initiator. Based on the above, 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 proposed change analyzes the NPSH for the Safety Injection and Containment Spray systems in a post-LOCA environment while operating in the recirculation mode with a debris laden fluid. No important to safety systems, structures, or components outside of the Safety Injection and Containment Spray systems are affected by the proposed change. The analysis concludes that adequate NPSH is maintained for the subject Safety Injection and Containment Spray system pumps such that system and component performance will meet the design bases criteria and core cooling acceptability maintained. Malfunction of a Safety Injection or Containment Spray component is not postulated as a result of the proposed change. Therefore, the proposed change does not 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.

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

   BASIS:

   The Safety Injection and Containment Spray systems are accident mitigating systems. The performed analysis concludes that these systems have adequate available NPSH and expected performance will remain unchanged. 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:

   The performed analysis concludes that the Safety Injection and Containment Spray Systems have adequate available NPSH and expected performance will remain unchanged. The consequences of malfunction of these systems are not impacted by the proposed change and do 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:

   The proposed change evaluates the performance of the Safety Injection and Containment Spray systems while operating in recirculation mode following a high energy line break in containment which produces fibrous and particulate debris that could potentially degrade the systems performance. The proposed change does not involve a physical change to the subject pumps or affect the manner of operation. The performed analysis concludes that the Safety Injection and Containment Spray Systems have adequate available NPSH and expected performance will remain unchanged. 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:

   The failure modes and effects tables for the Safety Injection and Containment Spray systems in the UFSAR were reviewed and remain valid. The analysis performed concludes the Safety Injection and Containment Spray Pumps have adequate NPSH margin and performance will be unchanged. Therefore the proposed change does not create 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.

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 performed analysis concludes that the Safety Injection and Containment Spray Systems have adequate available NPSH and expected performance will remain unchanged. The analysis for the Safety Injection Sump is based on two train operation and is bounded by fission product barrier analysis based on single train operation. The proposed change does not involve changes to a fission product barrier. Therefore, the proposed change 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:

UFSAR section 6.2.2.3.2.1 discusses the method for evaluating NPSH for the Containment Spray pumps and section 6.3.2.2.2.3 discusses the method for evaluating NPSH for the High Pressure Safety Injection pumps. The reduction in NPSH margin is a result of plant specific input parameter changes. The input parameters continue to be conservative with respect to the approved methodology. The methodology used for the subject pumps NPSH calculation is identical to that used in the current design basis calculation ECM07-001. 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

Proposed Change / Document: EC-41355, Evaluate Manual Operator Actions Outside Control Room For Certain Air Operated Valves after Accumulator is Exhausted

Description of Change:

Engineering Change (EC) 41355 clarifies the mission time and safety function of safety related motive gas accumulators and evaluates appropriate testing requirements of several manual local handwheels associated with Air Operated Valves (AOV), which may fail opposite their safe position upon exhausting the motive air in their associated accumulator in a postulated design basis event with an extended loss of Instrument Air (IA). EC-41355 also corrects Waterford 3 (WF3) Final Safety Analysis Report (FSAR) discrepancies identified in the section below. Specifically, this change adds information to the Design Basis Documents (DBD) and FSAR to clarify the mission time of the accumulators and to credit operator action outside the control room after the motive gas accumulators are exhausted.

Background:

NRC Generic Letter 88-14, Instrument Air System Supply Problems, directed licensees to:

1 Verify that maintenance practices, emergency procedures, and training are adequate to ensure that safety-related equipment will function as intended on loss of instrument air.
2 Verify that the design of the entire instrument air system including air or other pneumatic accumulators is in accordance with its intended function, including verification by test that air-operated safety-related components will perform as expected in accordance with all design-basis events, including a loss of the normal instrument air system. This design verification should include an analysis of current air operated component failure positions to verify that they are correct for assuring required safety functions.

During RF-18, CR-WF3-2012-6703 was initiated to identify that the manual handwheel override for ACC-112B was not able to change position of the valve following an actuator rebuild. CR-WF3-2012-6703 also identifies a design and licensing basis discrepancy regarding several other AOVs that may require repositioning after their air accumulator is exhausted. Currently, no testing is performed for several of the valves' manual handwheel functions, the valves are high above the floor. EC-41095 adds a backup air supply to Nitrogen Accumulator #1 & #2. A separate 10CFR50.59 evaluation was prepared for EC-41095. In addition, CR-WF3-2012-6703 identifies that the current license basis does not credit manual operator action for Component Cooling Water or Auxiliary Component Cooling Water valves that would require such action in order to place or maintain the valves in the position to support the safety function after safety related motive gas accumulators are exhausted.

Condition Identification (CI) 260438 identifies that the Nitrogen Accumulators are sized too small such that the ADVs can not operate for the 36 hours required in the FSAR. Potentially Reportable Event (PRE) 88-133 identifies that the accumulator also did not store enough gas for 10 hours and was therefore reportable. Corrective actions were taken to lower leakage such that 10 hours of operation was achievable.

PEIR 71128A (no associated 10CFR50.59 Evaluation) was performed in 1990 to provide specific accumulator accident durations or a "standard" 10 hour accumulator duration to allow time for operator action after a loss of instrument air. The background discussion in this PEIR provides a historical accounting of the development of the "standard" accumulator duration. Several followup actions were identified including 1) modifying the SI-602A(B) valve actuators, 2) testing to assure the accumulator systems meet minimum time requirements, and 3) investigate designs to recharge accumulators with high pressure gas cylinders. There was no follow-up action to develop

[1] 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.

specific procedures, training, or testing for using handwheel overrides in cases where the valve position change is required after the accumulator duration expires and there is no evidence of a 10CFR50.59 evaluation to determine acceptability of crediting manual operator action in place of what had been considered automatic.

PEIR 20106 revised the design and license basis documents to reflect 10 hour mission time for motive air on the ADVs, MS-116A(B), and EFW valves, EFW-223A(B), EFW-224A(B), EFW-228A(B), EFW-229A(B). Previously, the mission time was 24 - 36 hours, such that the valves did not require action outside the control room. Credit was taken in the revised design and license basis for manual local operation of the ADV and EFW valves, if necessary, during several design basis events. The 10CFR50.59 Evaluation addressed the possibility that manual operation would be necessary and accessibility of the valves and accumulators were also addressed. However, the evaluation did not provide specific detail about how the manual action would be accomplished. The evaluation also did not address testing or otherwise verifying the safety related function of the manual handwheel operators.

EC 15730 documents the 50.59 screening for ECS09-005, which was intended to replace the draft Safety Analysis DBD to provide the basis for the 10 hour motive gas accumulator mission time. However, ECS09-005 references to ECM89-002, ECM89-089, and W3-DBD-14 for the 10 hour mission time, which were referencing the Safety Analysis DBD for the basis for the 10 hour mission time. Currently ECS09-005 does not determine mission times, stay times, access routes, or other considerations to conclude that operator actions outside the control room following the accident will be possible. Therefore, there does not appear to be a properly documented basis for the 10 hour motive gas accumulator mission time.

The basis for a 10 hour mission time should be that either: 1) the safety function of the valve is complete after 10 hours; 2) the accumulator can be recharged before the 10 hour accumulator duration has elapsed; 3) the valve can be manually operated using a handwheel for the remainder of the valves mission time.

In the event of a loss of IA, the current license basis credits local handwheel operation of MS-116A(B), but does not explicitly credit handwheel operation of the other AOVs.

Specific Changes:

EC-41355 does not make any physical changes to the facility. Design and licensing basis documentation is updated to clarify the safety related function of the local manual handwheel on the following AOVs:
ACC-112A, ACC HEADER A TO ESSENTIAL CHILLERS ISOL
ACC-112B, ACC HEADER B TO ESSENTIAL CHILLERS ISOL
ACC-126A, ACC HEADER A CCW HX OUTL TEMPERATURE CONTROL
ACC-126B, ACC HEADER B CCW HX OUTL TEMPERATURE CONTROL
ACC-139A, ACC HEADER A RETURN FROM ESSENTIAL CHILLERS ISOL
ACC-139B, ACC HEADER B RETURN FROM ESSENTIAL CHILLERS ISOL
CC-114A, CCW PUMP A TO AB SUCTION CROSSCONNECT
CC-114B, CCW PUMP B TO AB SUCTION CROSSCONNECT
CC-115A, CCW PUMP AB TO A SUCTION CROSSCONNECT
CC-115B, CCW PUMP AB TO B SUCTION CROSSCONNECT
CC-126A, CCW PUMP A TO AB DISCHARGE CROSSCONNECT
CC-126B, CCW PUMP B TO AB DISCHARGE CROSSCONNECT
CC-127A, CCW PUMP AB TO A DISCHARGE CROSSCONNECT
CC-127B, CCW PUMP AB TO B DISCHARGE CROSSCONNECT
CC-200A, CCW HEADER A TO AB SUPPLY ISOLATION
CC-200B, CCW HEADER B TO AB SUPPLY ISOLATION
CC-301A, CCW HEADER A SUPPLY TO ESSENTIAL CHILLERS ISOL
CC-301B, CCW HEADER B SUPPLY TO ESSENTIAL CHILLERS ISOL
CC-322A, CCW HEADER A RETURN FROM ESSENTIAL CHILLERS ISOL
CC-322B, CCW HEADER B RETURN FROM ESSENTIAL CHILLERS ISOL
CC-563, CCW HEADER AB TO B RETURN ISOLATION
CC-620, FUEL POOL HEAT EXCH'S TEMPERATURE CONTROL

CC-727, CCW HEADER AB TO A RETURN ISOLATION
EFW-223A, EMERGENCY FEEDWATER HDR A TO SG1 BACKUP FLOW CNTRL
EFW-223B, EMERGENCY FEEDWATER HDR B TO SG2 BACKUP FLOW CNTRL
EFW-224A, EMERGENCY FEEDWATER HDR A TO SG1 PRIMARY FLOW CNTRL
EFW-224B, EMERGENCY FEEDWATER HDR B TO SG2 PRIMARY FLOW CNTRL
EFW-228A, EMERGENCY FEEDWATER HDR A TO SG1 BACKUP ISOLATION
EFW-228B, EMERGENCY FEEDWATER HDR B TO SG2 BACKUP ISOLATION
EFW-229A, EMERGENCY FEEDWATER HDR A TO SG1 PRIMARY ISOLATION
EFW-229B, EMERGENCY FEEDWATER HDR B TO SG2 PRIMARY ISOLATION
MS-116A, STEAM GENERATOR 1 ATMOSPHERE DUMP VALVE
MS-116B, STEAM GENERATOR 2 ATMOSPHERE DUMP VALVE

The existing remote control and local manual control functionality is not affected and either may be available depending on the availability of the IA system and or Nitrogen (NG) system following a design basis event. The existing remote control functionality is not affected when Instrument Air remains available. The existing design and license basis documents for the Emergency Feed Water (EFW) flow control and Atmospheric Dump Valves (ADV) already recognize the limited 10 hour duration of the various safety related IA and NG accumulators and credits the local manual handwheel function. The Licensing Basis for the Component Cooling Water (CCW) and Auxiliary Component Cooling Water (ACCW) does not explicitly identify a mission time for the AOV motive gas accumulators. This change adds information to the licensing basis to clarify the mission time of the accumulator and to credit operator action outside the control room after the motive gas accumulators are exhausted. This change also adds formal periodic testing of the manual handwheel function on several AOVs and updates the IA malfunction procedure to have guidance for when to operate the manual handwheel function following a design basis event.

Corrections and clarifications in the FSAR and in Design Basis Documents and calculations will also be made. For example, FSAR Table 9.2-4, Failure Modes and Effects Analysis for the CCWS and ACCWS Post LOCA, states that loss of instrument air has no affect because the active valves are provided with accumulators and/or springs to ensure proper alignment of the system. However, calculation ECM89-002 "Allowable Nitrogen Accumulator Leak Rate and ECM89-089 "allowable Instrument Air Accumulator Leak Rate" provide a leakage criteria based on a 10 hour mission time, and W3-DBD-014 "Safety Related Air Operated Valves Design Basis Document", states that manual handwheel positioning or refilling the nitrogen or air accumulators is required after 10 hours. ECS09-005, "Air Operated Valves – Design Basis Accident Times" was referenced in ECM89-002 for the basis of the selected mission time. However, ECS09-005 references back to ECM89-002 for the accumulator time and states that it does not establish the mission time. Therefore, a new evaluation must be performed to establish the basis for the accumulator mission time and to evaluate manual operator actions outside of the control room for valves that fail opposite their safe position after the motive gas accumulator is exhausted.

Specific Changes

EC-41355 does not make any physical changes to the facility. Design and licensing basis documentation is updated to clarify the safety related function of the local manual handwheel on the AOVs listed above. The basis for the existing 10 hour motive gas accumulator mission time is documented in calculation ECS09-005, "Air Operated Valves – Design Basis Accident Times". The basis includes credit for operator action outside the control room in place of automatic or remote manual operation. Since the proposed change evaluates a manual method of accomplishing design function, the change is being evaluated under the 50.59 rule.

FSAR Table 3.9-9

Added manual operator type in addition to the existing piston or diaphragm operator type for each of the subject valves except for the Essential Chiller coolant isolation valves.

Specific licensing basis changes are as follows:

{Added words are underlined and deleted words are struck through.}

FSAR 7.4.1.2 c) Redundancy and Diversity

The atmospheric dump valves are sized such that the reactor can be brought to hot standby assuming the loss of one valve. The control of these valves are safety related. In the event of loss of the non-safety related Instrument Air System, cooldown of the reactor to 350°F can then be accomplished through manual operation of the atmospheric dump valves. Each atmospheric dump valve has a handwheel which can be operated locally to override the actuator spring. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after the 10 hour mission time of the safety related Nitrogen Accumulator.

FSAR 7.4.2.3 Loss of Instrument Air Systems

Pneumatically operated valves in systems required for safe shutdown are designed to fail in the position required for safe system operation in the plant shutdown mode, except for the atmospheric dump valves which are fail closed and the CCW Cross-Connect Valves, which fail open. Valves which are in required flow paths will fail open on loss of instrument air. The CCW Cross-Connect valves may be gagged closed by local manual means in the event of loss of air. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after 10 hour mission time of the safety related motive gas accumulator. The atmospheric dump valves may be opened by local manual means in the event of loss of air. Valves which isolate nonessential portions of the system from portions required for safe shutdown are designed to fail close. Valve failure positions are shown on the systems P&I diagrams. The pressurizer spray valves and auxiliary spray valves fail closed on loss of instrument air. Pressurizer pressure is controlled by operation of the electric pressurizer heaters, and over pressure is relieved by pressurizer safety valves. The valves in the charging line of the CVCS fail open. The loss of instrument air system will not preclude the safe shutdown of the plant.

FSAR 9.3.1.1 Design Bases

The compressed air system, consisting of the Instrument and Service Air Systems, is designed to provide a reliable supply of dry, oil-free air for pneumatic instruments and controls, pneumatically operated valves and the necessary service air for normal plant operation and maintenance. The system serves no safety function since it is not required to achieve safe shutdown or to mitigate the consequences of an accident. Air accumulators are provided on valves where instrument air is required for operation during the safe shutdown of the plant following an accident or to mitigate the consequences of an accident. Air accumulators required for containment isolation are provided with safety related remote makeup capability to ensure long term operability of the valves following loss of the plant instrument air system. Other safety related air accumulators are capable of providing motive air to pneumatically operated valves for 10 hours. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after 10 hours. Safety-related valves with air accumulators are listed in Table 9.3-1a and with nitrogen accumulators in Table 9.3-1b.

FSAR 9.3.6.3.3 A cooldown analysis was done to determine how much time would be required to bring the plant from power operation to shutdown cooling conditions. The analysis was done to meet the requirements of BTP RSB 5-1. Only safety grade systems were used. Offsite power was assumed to be lost and the most limiting single failure, one atmospheric dump valve fails to open, was assumed. Thus the cooldown was accomplished using only one steam generator since no credit for the cross tie piping was taken. As shown in Figures 9.3-8a and 9.3-8b, shutdown cooling conditions were reached in Less than ten hours, when both hot leg temperatures are reduced to 400°F. This temperature is the design temperature of SDCS components. A ten hour minimum backup supply of motive gas for the atmospheric dump valve actuators (reference subsection 10.2.1) is provided by Safety Class 3, Seismic Category I accumulators to assure the valves remain operable from the control room until shutdown cooling entry conditions are satisfied. The maximum operating temperature of the SDCS is 350°F, therefore it is desirable to achieve ≤ 350°F prior to initiating SDC. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after 10 hours.

FSAR 9.3.9.1 Design Bases

The nitrogen system is designed to provide a reliable supply of nitrogen for the nitrogen accumulators on pneumatically operated valves and the necessary nitrogen for normal plant operation and maintenance. Nitrogen accumulators are provided as a backup source of compressed gas for various safety-related valves

needed to mitigate the consequences of an accident or for the safe shutdown of the plant following an accident. Safety-related valves serviced by nitrogen accumulators are listed in Table 9.3-1b. Safety related Nitrogen accumulators are capable of providing motive air to pneumatically operated valves for 10 hours. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after 10 hours.

FSAR 9.3.9.3 Safety Evaluation

A complete loss of the nitrogen supply system during full power operation does not reduce the ability of the reactor protective system or the engineered safety features and their supporting systems to safely shutdown the reactor or to mitigate the consequences of an accident. The nitrogen supply system exclusive of the nitrogen accumulators is a non-safety-related system, serves no safety function and is not designed to seismic requirements. The portion of the nitrogen piping and valves penetrating the containment is designed to Safety Class 2 and seismic Category I requirements (refer to Subsection 6.2.4). The containment nitrogen header outer isolation valve is designed to fail closed. The nitrogen accumulators provide a backup source of compressed gas to operate pneumatic safetyrelated valves in the event of a loss of instrument air. Table 9.3-lb lists all safety-related valves provided with backup nitrogen accumulators. The accumulators are designed to Safety Class 3 and seismic Category I requirements. They are sized to permit sufficient stroking of all specified valves in the course of performing their safety-related functions. Upon a loss of instrument air, a low pressure signal will open the respective nitrogen supply valves. Nitrogen that is stored in the accumulators will then charge the valve operators' supply lines permitting continued valve operation. Safety related Nitrogen accumulators are capable of providing motive air to pneumatically operated valves for 10 hours. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after 10 hours.

FSAR Table 9.2-4 Failure Modes and Effects Analysis for the CCWS and ACCWS Post LOCA

Failure Mode	Effect on System	Remarks
Loss of air	No effect	The active valves are provided with accumulators and / or springs to ensure proper alignment of the system. Safety related accumulatorsare capable of providing motive air to pneumatically operated valves for 10 hours. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after 10 hours.

FSAR 10.3.1

Current version of 10.3.1 implemented by PEIR 20106 in 1991 – Seismic Category I accumulators, sized for a minimum of 10 hours operation, are provided for the atmospheric dump valves, and associated EFS valves. Ten hours is consistent with the natural circulation cooldown analysis described in Section 9.3.6.3.3. The valves can also be manually operated if required. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after the 10 hour mission time of the safety related Nitrogen Accumulator. Main steam isolation valves operators are hydraulic-pneumantic.

Old version of 10.3.1 - Safety valves and power-operated relief (atmospheric dump) valves are provided for each steam generator immediately outside the containment structure upstream of the main steam isolation valves. The power-operated relief valves (one per steam generator) are air operated and fail in the closed position on loss of air supply. Backup seismic Category I air accumulators are provided for these valves in order to maintain their operability manually from the control room. The backup air supply is sized to maintain the valves operable for 36 hr. The power-operated relief valves are also equipped with handwheels for local manual operation. Thus, the requirements of GDC 34, "Residual Heat Removal," are satisfied.

GDC 34— Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

FSAR 10.3.3 EVALUATION

Heat dissipation requirements during plant startup, hot shutdown, and cooldown are normally met by bypassing steam to the condenser via the turbine bypass system described in Subsection 10.3.1 and 10.4.4. If the bypass system is not available, the atmospheric dump valves are adequately sized (each valve is sized for five percent of the total throttle flow) to remove decay heat during plant cooldowns. In addition, the Atmospheric Dump Valve (ADV) is also credited with safety related function of providing decay heat removal during a small break LOCA event. (Refer to Section 6.3) The 60 percent capacity (of total throttle flow) SBS permits full load rejection, to house load, without reactor trip and without lifting of main steam safety valves. Failure of the SBS to function during turbine-reactor power mismatch will result in a reactor trip, causing the main steam pressure to rise. However, the main steam safety valves, which are adequately sized to permit load rejection from full power, will open and prevent pressure rise above 110 percent of the maximum allowable pressure for the steam generator.

FSAR 10.4.9.3.1 Pipe Break

The EFS is used only for emergency shutdown of the reactor when the Main Feedwater System is inoperative. Accordingly, based on recommendations of Branch Technical Position APCSB3-1 (as further discussed in Subsection 10.4.9.1), it is classified as a moderate energy system and high energy pipe breaks are not postulated.

The EFS isolation valves and TSSV are located adjacent to the break exclusion area portions of the main steam and feedwater piping. Accordingly, pipe rupture effects of the main steam and feedwater piping on the EFS isolating and TSSV are not considered.

In the event of a main steam or feedwater line break there are four EFS isolation valves in the lines to each steam generator, arranged in two parallel paths, as indicated in Figure 10.4-6. The valves are pneumatically operated, fail open, and provided with accumulators sized for a minimum of 10 hours EFS operation. The valves are also equipped with handwheels for local manual operation. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after 10 hour mission time of the safety related motive gas accumulator.

FSAR Table 10.4-14, Emergency Feedwater System Failure Mode and Effects Analysis

Failure Mode	Effect on System	Method of Detection	Monitor	Remarks
Loss of air	EFW Flow Control Valves fail open.	Steam Gen. Level	Control Room Indication	After 10 hour mission time of the safety related motive gas accumulator, procedures are established for operating the manual handwheel overrides.

FSAR Table 10.4.9A-1 Acceptance Criterion 4

General Design Criterion 19, as related to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown.

The EFS is automatically initiated by an Emergency Feedwater Actuation Signal (EFAS) as described in Section 7.3. The EFS can also be started manually from the Main Control Room and the auxiliary control panel. Safety related display information located in the main control room provides the operator with sufficient information to perform the required safety functions. See Section 7.5. Procedures are established for operating manual handwheel overrides or lining up backup air supplies for continued safety function after 10 hour mission time of the safety related motive gas accumulator.

FSAR Table 10.4.9A-2

Concern - Some plants require local manual realignment of valves to conduct periodic pump surveillance tests on one AFW system train. When such plants are in this test mode and there is only one remaining AFW system train available to respond to a demand for initiation of AFW system operation, the AFW system redundancy and ability to withstand a single failure are lost.

Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system from the test mode to its operational alignment.

Response: ~~Not applicable. Local manual realignment of valves to conduct periodic pump surveillance tests on EFS trains is not required.~~ Procedures are established for testing the manual handwheel overrides on the EFS Flow Control Valves. The procedure performs a single stroke and then restores the valve to the operational alignment. Other than this brief manual realignment, local manual realignment of valves to conduct surveillance tests on EFS trains is not required. Periodic pump surveillance tests can be conducted by merely starting the pump and water will be pumped from the CSP through the mini-flow recirculation lines back to the CSP. ~~There is in fact no test on any EFS system active component which will reduce the availability of the EFS system flow train.~~ The manual handwheel override test will have specific instructions regarding the need to immediately restore the handwheel position to normal upon completion or upon any interruption of the test.

Reasoning:

This 50.59 Evaluation is performed by answering the eight questions in section II with the following additional considerations addressing the potentially adverse impact of the change to the operator action required to manually position certain AOVs during postuated events with an extended loss of instrument air.

In the event of an extended loss of instrument air, where the safety related accumulators are exhausted, manual operator action is credited outside the control room in order to either gag valves in their safe position prior to exhausting the accumulators, line up safety related backup air supplies for certain accumulators, or reposition or maneuver control valves using manual handwheel overrides. Calculation ECS09-005 is updated and an evaluation is performed herein, in accordance with NRC Information Notice 97-78, to demonstrate that required operator actions are achievable. Post Return to Service Actions are established to track procedure changes and training necessary to protect the assumptions in the analysis. New testing requirements will be established to demonstrate the capability of the manual handwheel override function credited in the analysis. The existing remote control functionality is not affected when Instrument Air remains available.

Technical Specification (TS) 3.7.1.2 requires three emergency feedwater (EFW) pumps and two flow paths shall be OPERABLE in modes 1, 2, and 3.

TS 3.7.1.7 requires each Atmospheric Dump Valve (ADV) shall be OPERABLE in modes 1, 2, 3, and 4.

TS 3.7.3 requires at least two independent component cooling water and associated auxiliary component cooling water trains shall be OPERABLE.

TS 3.7.4 requires two Independent trains of ultimate heat sink (UHS) cooling towers shall be OPERABLE with each train consisting of a dry cooling tower (DCT) and a wet mechanical draft cooling tower (WCT)

The TS definition of OPERABLE/OPERABILITY is a system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

If an operator action must be performed prior to a system being capable of performing its specified safety function, then it must be evaluated with respect to the guidance presented in NRC Information Notice 97-78,

Crediting of Operator Actions In Place of Automatic Actions and Modifications of Operator Actions, Including Response Times, October 23, 1997, NRC Regulatory Issue Summary 2005-20, Revision To NRC Inspection Manual Part 9900 Technical Guidance, Operability Determinations & Functionality Assessments For Resolution Of Degraded Or Nonconforming Conditions Adverse To Quality Or Safety, April 16, 2008, and NRC Inspection Manual Technical Guidance Part 9900 ITSB, Operability Determinations & Functionality Assessments For Resolution Of Degraded Or Nonconforming Conditions Adverse To Quality Or Safety.

NRC INFORMATION NOTICE 97-78 lists the following requirements for crediting an operator action:

> The original design of nuclear power plant safety systems and their ability to respond to design-basis accidents were described in licensees' FSARs and were reviewed and approved by the NRC. Most safety systems were designed to rely on automatic system actuation to ensure that the safety systems were capable of carrying out their intended functions. In a few cases, limited operator actions, when appropriately justified, were approved. Proposed changes that substitute manual action for automatic system actuation or modify existing operator actions, including operator response times, previously reviewed and approved during the original licensing review of the plant will, in all likelihood, raise the possibility of a USQ. Such changes must be evaluated under the criteria of 10 CFR 50.59 to determine whether a USQ is involved and whether NRC review and approval is required before implementation. A licensee may not make such changes before it receives approval from the NRC when the change, test, or experiment may (1) increase the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously analyzed in the FSAR, (2) create the possibility of an accident or a malfunction of a different type than any previously evaluated in the FSAR, or (3) reduce the margin of safety as defined in the basis for any TS. In the NRC staff's experience, many of the changes of the type described above proposed by licensees do involve a USQ.

NRC INFORMATION NOTICE 97-78 also lists specific requirements the NRC will use to review new operator actions. Based on these guidelines, the NRC's reviews of licensees' analyses typically include, but are not limited to, (1) the specific operator actions required; (2) the potentially harsh or inhospitable environmental conditions expected; (3) a general discussion of the ingress/egress paths taken by the operators to accomplish functions; (4) the procedural guidance for required actions; (5) the specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions; (6) any additional support personnel and/or equipment required by the operator to carry out actions; (7) a description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken; (8) the ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery; and (9) consideration of the risk significance of the proposed operator actions.

NRC REGULATORY ISSUE SUMMARY 2005-20 issued a new version of NRC Inspection Manual Technical Guidance Part 9900 ITSB. The NRC Inspection Manual lists the following requirements for crediting an operator action:

> For situations where substitution of manual action for automatic action is proposed for an operability determination, the evaluation of manual action must focus on the physical differences between automatic and manual action and the ability of the manual action to accomplish the specified safety function or functions. The physical differences to be considered include the ability to recognize input signals for action,

ready access to or recognition of setpoints, design nuances that may complicate subsequent manual operation (such as auto-reset, repositioning on temperature or pressure), timing required for automatic action, minimum staffing requirements, and emergency operating procedures written for the automatic mode of operation. The licensee should have written procedures in place and personnel should be trained on the procedures before any manual action is substituted for the loss of an automatic action.

The assignment of a dedicated operator for a manual action requires written procedures and full consideration of all pertinent differences. The consideration of a manual action in remote areas must include the abilities of the assigned personnel and how much time is needed to reach the area, training of personnel to accomplish the task, and occupational hazards such as radiation, temperature, chemical, sound, or visibility hazards. One reasonable test of the reliability and effectiveness of a manual action may be the approval of the manual action for the same function at a similar facility.

The manual operator action was evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994, American National Standard Time Response Design Criteria for Safety Related Operator Actions, August 23, 1994, and is addressed in Question #2.

Summary of Evaluation:

This evaluation determined that clarifying the safety function and mission time of the accumulators, evaluating appropriate testing requirements of several manual local handwheels associated with Air Operated Valves (AOV), and crediting operator action outside the control room after the motive gas accumulators are exhausted do not represent any unreviewed safety question and do not require prior NRC review and approval.

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: Dale V. Gallodoro / See EC-41355 / EOI / Design Engineering / See EC-41355
Name (print) / Signature / Company / Department / Date

Reviewer: Nasser Pazooki / See EC-41355 / EOI / Nuclear Analysis Engineering / 12/29/12
Name (print) / Signature / Company / Department / Date

Reviewer: Marc McCloskey / See EC-41355 / EOI / Design Engineering / 12-29-12
Name (print) / Signature / Company / Department / Date

OSRC: Keith Nichols / 12/30/12
Chairman's Name (print) / Signature / Date

W3 12-27
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: EC 41355 CLARIFIES THE MISSION TIME AND SAFETY FUNCTION OF SAFETY RELATED MOTIVE GAS ACCUMULATORS AND EVALUATES APPROPRIATE TESTING REQUIREMENTS OF SEVERAL MANUAL LOCAL HANDWHEEL OVERRIDES ASSOCIATED WITH AIR OPERATED VALVES (AOV) LISTED IN THE DESCRIPTION OF CHANGE SECTION ABOVE, WHICH MAY FAIL OPPOSITE THEIR SAFE POSITION UPON EXHAUSTING THE MOTIVE GAS IN THEIR ASSOCIATED ACCUMULATOR IN A POSTULATED DESIGN BASIS EVENT WITH AN EXTENDED LOSS OF IA. THE CHAPTER 15 ACCIDENTS EVALUATED IN THE UFSAR THAT MAY BE AFFECTED BY THE OPERATION OF THE SUBJECT VALVES ARE LOSS OF COOLANT ACCIDENT (LOCA) (FSAR 15.6.3.3) AND OTHER EVENTS THAT REQUIRE COMPONENT COOLING AND COULD REQUIRE EMERGENCY FEEDWATER AND STEAM GENERATOR COOLDOWN. THE INSTRUMENT AIR SYSTEM AND NITROGEN SYSTEM ARE NOT INITIATORS OF ANY ACCIDENT PREVIOUSLY EVALUATED IN THE FSAR. EACH OF THE SUBJECT VALVES IS NORMALLY REMOTELY OPERATED IN AUTOMATIC USING NORMAL INSTRUMENT AIR AS THE MOTIVE GAS. SAFETY RELATED MOTIVE GAS ACCUMULATORS ARE SIZED TO PROVIDE 10 HOURS OF VALVE OPERATION AFTER A LOSS OF NORMAL INSTRUMENT AIR. THE EXISTING IA MALFUNCTION PROCEDURE REQUIRES ACTIONS TO ENTER TS ACTION STATEMENTS PRIOR TO REACHING THE MINIMUM REQUIRED AIR PRESSURE IN THE ACCUMULATORS THAT ENSURES THE DESIGN BASIS 10 HOUR MISSION TIME. THE EXISTING DESIGN BASIS CREDITS OPERATOR ACTIONS OUTSIDE THE CONTROL ROOM FOR ALIGNING BACKUP AIR SUPPLIES AND USING MANUAL HANDWHEEL OVERRIDES, ALL OF WHICH ARE EVALUATED AS BEING ACCESSIBLE. OPENING OR CLOSING THE VALVES AFFECTED BY THIS CHANGE IS NOT AN ACCIDENT INITIATOR. THE OPERATOR ACTION WILL BE IN RESPONSE TO LOSS OF INSTRUMENT AIR THAT HAS AN EXISTING PLANT ANALYSIS RESPONSE THAT REMAINS UNCHANGED. NO PHYSICAL CHANGES TO THE PLANT ARE AUTHORIZED BY THIS EC. 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: EC 41355 CLARIFIES THE MISSION TIME AND SAFETY FUNCTION OF SAFETY RELATED MOTIVE GAS ACCUMULATORS AND EVALUATES APPROPRIATE TESTING REQUIREMENTS OF SEVERAL MANUAL LOCAL HANDWHEEL OVERRIDES ASSOCIATED WITH AIR OPERATED VALVES (AOV) LISTED IN THE DESCRIPTION OF CHANGE SECTION ABOVE, WHICH MAY FAIL OPPOSITE THEIR SAFE POSITION UPON EXHAUSTING THE MOTIVE AIR IN THEIR ASSOCIATED ACCUMULATOR IN A POSTULATED DESIGN BASIS EVENT WITH AN EXTENDED LOSS OF IA. EACH OF THE SUBJECT VALVES IS NORMALLY REMOTELY OPERATED IN AUTOMATIC USING NORMAL INSTRUMENT AIR AS THE MOTIVE GAS. SAFETY RELATED MOTIVE GAS ACCUMULATORS ARE SIZED TO PROVIDE 10 HOURS OF VALVE OPERATION AFTER A LOSS OF NORMAL INSTRUMENT AIR. THE EXISTING BACKUP AIR SUPPLIES OR MANUAL HANDWHEEL OVERRIDES CREDITED AFTER 10 HOURS WERE DESIGNED AND INSTALLED TO MEET THE SAME STANDARDS AND SEISMIC QUALIFICATION AS THE PNEUMATIC ACTUATOR. THE EXISTING IA MALFUNCTION PROCEDURE REQUIRES ACTIONS TO ENTER TS ACTION STATEMENTS PRIOR TO REACHING THE MINIMUM REQUIRED AIR PRESSURE IN THE ACCUMULATORS THAT ENSURES THE DESIGN BASIS 10 HOUR MISSION TIME. THE EXISTING DESIGN BASIS CREDITS OPERATOR ACTIONS OUTSIDE THE CONTROL ROOM FOR ALIGNING BACKUP AIR SUPPLIES AND USING MANUAL HANDWHEEL OVERRIDES, ALL OF WHICH ARE EVALUATED AS BEING ACCESSIBLE. THE MANUAL ACTION CAN BE ACCOMPLISHED PRIOR TO THE 10 HOUR TIME CAPACITY OF THE SAFETY RELATED MOTIVE GAS ACCUMULATORS. A CREDIBLE ERROR IS UNLIKELY SINCE IN MOST CASES THE ACTION IS SIMPLY TO GAG A VALVE IN ITS SAFE POSITION OR LINE UP A BACKUP AIR SUPPLY TO THE SAFETY RELATED MOTIVE GAS ACCUMULATORS. IN THE CASE OF THE EMERGENCY FEEDWATER (EFW) CONTROL VALVES OR ATMOSPHERIC DUMP VALVES (ADV), OPERATORS WILL BE IN CONSTANT COMMUNICATION WITH THE CONTROL ROOM VIA RADIOS AND REACTING TO CONTROL ROOM INSTRUCTIONS TO OPEN AND CLOSE THE VALVES TO CONTROL REACTOR COOLANT TEMPERATURE AND STEAM

GENERATOR LEVEL. THE INHERENT COMPENSATING PROVISION REMAINS THE REDUNDANT TRAIN WITH THE PROPOSED CHANGE. OPERATOR ACTION IS CREDITED TO EITHER LINE UP BACKUP AIR SUPPLIES OR OPERATE HANDWHEEL OVERRIDES TO ALLOW OPERATION OF VALVES AFTER THE 10 HOUR MOTIVE GAS SUPPLY IS EXHAUSTED. IN EACH CASE THE DESIGN MAINTAINS QUALIFICATION, SEPARATION, AND REDUNDANCY OF EQUIPMENT.

THE MANUAL OPERATOR ACTION IS EVALUATED AGAINST NRC INFORMATION NOTICE 97-78, NRC REGULATORY ISSUE SUMMARY 2005-20, AND ANSI/ANS-58.8-1994 ON SYSTEM OPERATION POST RECIRCULATION ACTUATION SIGNAL (RAS). THE TEN PRIMARY ATTRIBUTE EVALUATIONS ARE SPECIFICALLY LISTED BELOW.

(1) THE SPECIFIC OPERATOR ACTIONS REQUIRED;

THE OPERATOR ACTIONS REQUIRED TO COMPENSATE FOR AN INSTRUMENT AIR MALFUNCTION IS CONTAINED IN PROCEDURE OP-901-511. POST RETURN TO SERVICE ACTIONS FOR EC-41355 ARE ESTABLISHED TO TRACK PROCEDURE CHANGES AND TRAINING NECESSARY TO PROTECT THE ASSUMPTIONS IN THE ANALYSIS. THE SPECIFIC CHANGE IN OPERATOR ACTION IS TO:

ACC-126A(B) – CCW SUPPLY TEMPERATURE CONTROL VALVES

ACC-126A(B) THROTTLING USING THE LOCAL HANDWHEEL OVERRIDE MUST BE PERFORMED PERIODICALLY AFTER 10 HOURS DURING AN EVENT TO MAINTAIN CCW SUPPLY TEMPERATURE CONSISTENT WITH THE PARTICULAR EVENT ANALYSIS ASSUMPTIONS TO PRESERVE WATER INVENTORY. FOR EXAMPLE, FOR EVENTS ASSOCIATED WITH A SAFETY INJECTION ACTUATION SIGNAL (SIAS), CCW SUPPLY TEMPERATURE MUST BE CONTROLLED AT 115°F. ALL NUCLEAR AUXILIARY OPERATORS CARRY PORTABLE RADIOS FOR CONTINUOUS COMMUNICATION WITH THE CONTROL ROOM.

CC-114 A(B), CC-115 A(B), CC-126 A(B), CC-127 A(B) CCW PUMP ISOLATION VALVES

PRIOR TO 10 HOURS, WHILE THE ACCUMULATOR PRESSURE IS STILL ADEQUATE TO HOLD THE CCW CROSS-CONNECT VALVES IN THE CLOSED POSITION, OPERATORS MUST BE DISPATCHED TO GAG THE VALVES THAT WERE CLOSED BY THE SIAS SUCH THAT THEY WILL NOT REOPEN WHEN THE ACCUMULATOR IS EXHAUSTED. BASED ON G160 SH2, ONLY ONE OF THE TWO VALVES IN EACH LEG NEED TO BE CLOSED TO PROVIDE TRAIN SEPARATION. THIS MAY BE DONE BY ONLY ENTERING THE AB TRAIN CCW PUMP ROOM AND GAGGING AT MOST TWO VALVES. NOTE THAT THE TWO VALVES TO BE GAGGED WILL DEPEND ON THE ALIGNMENT OF THE AB CCW PUMP. ALL NUCLEAR AUXILIARY OPERATORS CARRY PORTABLE RADIOS FOR CONTINUOUS COMMUNICATION WITH THE CONTROL ROOM.

CC-200 A(B), CC-563, CC-727 CCW Train A and B Isolation Valves

PRIOR TO 10 HOURS, WHILE THE ACCUMULATOR PRESSURE IS STILL ADEQUATE TO HOLD THE CCW NON-ESSENTIAL HEADER ISOLATION VALVES IN THE OPEN POSITION, OPERATORS MUST MANUALLY LINE UP FUEL POOL COOLING USING REMAINING MOTIVE GAS IN THE ACCUMULATORS AND THEN IMMEDIATELY GAG THE VALVES OPEN SUCH THAT THEY WILL NOT CLOSE WHEN THE ACCUMULATOR IS EXHAUSTED. ONLY ONE OF THE TWO TRAINS VALVES (TWO VALVES) NEED TO BE OPEN TO PROVIDE FLOW TO THE NON-ESSENTIAL LOOP FOR FUEL POOL COOLING. THE OPERATOR MUST MAKE ADJUSTMENTS BASED ON CONTROL ROOM INDICATION AND DIRECTION TO AVOID INADVERTENTLY ISOLATING EQUIPMENT. ALL NUCLEAR AUXILIARY OPERATORS CARRY PORTABLE RADIOS FOR CONTINUOUS COMMUNICATION WITH THE CONTROL ROOM.

CC-301A(B), CC-322A(B), ACC-112A(B), ACC-139A(B) CCW/ACCW Supply to Chillers Isolation Valves

A BACKUP AIR SUPPLY CAPABLE OF RECHARGING NITROGEN ACCUMULATOR I(II) TO ALLOW ONE STROKE OF THE CHILLER COOLANT SELECT VALVES IS PROVIDED THAT NEEDS TO BE LINED UP JUST PRIOR TO SWITCHING FROM WET MODE TO DRY MODE COOLING SEVERAL DAYS INTO THE EVENT. THIS SWITCHING IS DONE WHEN OPERATORS AND ENGINEERS ARE CONFIDENT THAT CCW TEMPERATURE CAN BE MAINTAINED LESS THAN 110°F LONG TERM WITH ONLY THE DCT BY

MANEUVERING THREE VALVES PER TRAIN (IA-9542, IA-9543, AND IA-9544 (IA9552, IA-9553, AND IA-9554)).

CC-620 Fuel Pool Temperature Control Valve

WHEN THE NON-ESSENTIAL CCW LOOP IS RESTORED (SEE CC-200 A(B), CC-563, CC-727 ABOVE), CC-620 MUST BE THROTTLED OPEN TO PROVIDE APPROXIMATELY 500 GPM IN ACCORDANCE WITH CALCULATION MNQ9-2, COMPONENT COOLING WATER SYSTEM FLOW. THE OPERATOR MUST MAKE ADJUSTMENTS BASED ON CONTROL ROOM INDICATION AND DIRECTION TO AVOID INADVERTENTLY ISOLATING EQUIPMENT. ALL NUCLEAR AUXILIARY OPERATORS CARRY PORTABLE RADIOS FOR CONTINUOUS COMMUNICATION WITH THE CONTROL ROOM.

MS-116 A(B) MAIN STEAM ATMOSPHERIC DUMP VALVES; EFW-223 A(B), EFW-224A(B) EFW CONTROL VALVES; EFW-228 A(B), EFW-229 A(B), EFW ISOLATION VALVES

FOR EVENTS WHERE REACHING SDC ENTRY CONDITIONS TAKES LONGER THAN 10 HOURS, EFW FLOW CONTROL VALVES, EFW-223A(B) AND EFW-224A(B), AND MS-116 A(B) MAIN STEAM ATMOSPHERIC DUMP VALVES, MUST BE MANUALLY CONTROLLED TO CONTINUE TO PROVIDE REACTOR COOLANT SYSTEM COOLDOWN AND TO MAINTAIN STEAM GENERATOR LEVEL WITHIN THE ALLOWABLE BAND. IF FOR SOME REASON EFW-223A(B) OR EFW-224A(B) CANNOT BE CLOSED, THEN THE ASSOCIATED EFW ISOLATION VALVE, EFW-228A(B) / EFW-229A(B) MUST BE CLOSED. THE OPERATOR MUST BE IN CONSTANT COMMUNICATION WITH THE CONTROL ROOM AND MUST MAKE ADJUSTMENTS BASED ON CONTROL ROOM INDICATION AND DIRECTION TO AVOID INADVERTENTLY ISOLATING EQUIPMENT. ALL NUCLEAR AUXILIARY OPERATORS CARRY PORTABLE RADIOS FOR CONTINUOUS COMMUNICATION WITH THE CONTROL ROOM.

(2) THE POTENTIALLY HARSH OR INHOSPITABLE ENVIRONMENTAL CONDITIONS EXPECTED;

THE EVENTS OF INTEREST IS A LOSS OF COOLANT ACCIDENT (LOCA) (FSAR 15.6.3.3) AND OTHER EVENTS THAT REQUIRE COMPONENT COOLING AND COULD REQUIRE EMERGENCY FEEDWATER AND STEAM GENERATOR COOLDOWN. THE ANALYSIS OF RECORD FOR EVENTS WITH DOSE CONSEQUENCES CONCLUDES THAT SHUTDOWN COOLING ENTRY CONDITIONS ARE REACHED PRIOR TO 8 HOURS, WHICH IS WITHIN THE DURATION OF THE SAFETY RELATED MOTIVE GAS ACCUMULATORS. THEREFORE, NO LOCAL OPERATION OF THE ADV OR EFW CONTROL VALVES IS REQUIRED FOR ACCIDENTS WITH DOSE CONSEQUENCES. THE ANALYSIS IN ECS09-005 DEMONSTRATES THAT THE EFW CONTROL VALVES AND ADV CAN BE OPERATED WITH THE HANDWHEEL OVERRIDE FOR EVENTS THAT TAKE LONGER THAN 10 HOURS TO REACH SDC ENTRY CONDITIONS (TORNADO AND NATURAL CIRCULATION COOLDOWN WITH FAILED ADV). FOR THE AFFECTED ACCW AND CCW VALVES, THE ECS09-005 ANALYSIS DEMONSTRATES THAT THE ENVIRONMENTAL CONDITIONS ALLOW SUFFICIENT TIME TO PERFORM THE REQUIRED LOCAL ACTIONS WITHOUT EXCEEDING ESTABLISHED DOSE LIMITS. THEREFORE, THERE IS NO ADVERSE IMPACT.

(3) A GENERAL DISCUSSION OF THE INGRESS/EGRESS PATHS TAKEN BY THE OPERATORS TO ACCOMPLISH FUNCTIONS;

THE MAJORITY OF THE OP-901-511 ACTIONS OCCUR FROM THE CONTROL ROOM. EXISTING ACTIONS INCLUDE MANUAL LOCAL OPERATOR ACTION TO ALIGN EIA, WHICH IS NEAR THE SUBJECT ACCW AND CCW VALVES. FOR DESIGN BASIS EVENTS THAT MAY REQUIRE HANDWHEEL OVERRIDE OPERATION OF EFW CONTROL VALVES OR ADVS, THERE ARE NO DOSE CONSEQUENCES AND SO ALL AREAS ARE ACCESSIBLE. EMERGENCY LIGHTING IS AVAILABLE IN ALL LOCATIONS NEEDED AND THE ENVIRONMENT CONDITIONS WILL NOT BE ADVERSE, AS DISCUSED IN (2). ALL NUCLEAR AUXILIARY OPERATORS CARRY FLASHLIGHTS WHICH WILL PROVIDE ADEQUATE LIGHTING FOR INGRESS AND EGRESS TO THE VALVE. LADDERS ARE AVAILABLE IN DESIGNATED STORAGE LOCATIONS IN THE PLANT TO ALLOW ACCESS TO VALVES HIGH OFF THE FLOOR. SCAFFOLDING IS CURRENTLY IN PLACE AND WILL BE MADE PERMANENT. POST RETURN TO SERVICE ACTIONS FOR EC-41355 ARE ESTABLISHED TO TRACK PROCEDURE CHANGES, TRAINING, AND OTHER ITEMS NECESSARY TO PROTECT THE ASSUMPTIONS IN THE ANALYSIS. THEREFORE, THERE IS NO ADVERSE IMPACT.

(4) THE PROCEDURAL GUIDANCE FOR REQUIRED ACTIONS;

THE OPERATOR ACTIONS REQUIRED TO COMPENSATE FOR AN INSTRUMENT AIR MALFUNCTION IS CONTAINED IN PROCEDURE OP-901-511. POST RETURN TO SERVICE ACTIONS FOR EC-41355 ARE ESTABLISHED TO TRACK

PROCEDURE CHANGES AND TRAINING NECESSARY TO PROTECT THE ASSUMPTIONS IN THE ANALYSIS. THE SPECIFIC CHANGE IN OPERATOR ACTIONS ARE LISTED IN (1) ABOVE . NONE OF THE MANUAL ACTIONS WOULD BE REQUIRED EXCEPT IN THE EVENT OF A LOSS OF INSTRUMENT AIR. THE ORIGINAL REMOTE MANUAL ACTIONS AND LOCAL PNEUMATIC CONTROL MANUAL ACTIONS PREVIOUSLY EVALUATED WOULD BE AVAILABLE EXCEPT IN THE EVENT OF A LOSS OF IA SYSTEM. THEREFORE, THERE IS NO ADVERSE IMPACT.

(5) THE SPECIFIC OPERATOR TRAINING NECESSARY TO CARRY OUT ACTIONS, INCLUDING ANY OPERATOR QUALIFICATIONS REQUIRED TO CARRY OUT ACTIONS;

TRAINING ON THE ACTIONS TO RESPOND TO AN IA MALFUNCTION IS PART OF OPERATOR REQUALIFICATION TRAINING. THE ACTIONS ARE RELATIVELY STRAIGHFORWARD TO EXPERIENCED OPERATIONS PERSONNEL, INVOLVING THE POSITIONING AND GAGGING OF VALVES, WHICH ARE ROUTINELY PERFORMED. IN ADDITION, EN-HU-104, HUMAN PERFORMANCE PROGRAM, INDICATES IT IS STANDARD PRACTICE TO BRIEF THE OPERATORS DEDICATED TO PERFORMING SPECIFIC ACTIONS AS TO COMMUNICATIONS PROTOCOLS, STANDBY AREAS, TRAVEL ROUTES, TIME LIMITS, THE REQUIRED PROCEDURE STEPS, AND ACTIONS REQUIRED JUST PRIOR TO PERFORMING THE REQUIRED ACTION. THE STANDARD PRACTICE WOULD ALSO APPLY FOR A POTENTIAL IA MALFUNCTION. THEREFORE, THERE IS NO ADVERSE IMPACT.

(6) ANY ADDITIONAL SUPPORT PERSONNEL AND/OR EQUIPMENT REQUIRED BY THE OPERATOR TO CARRY OUT ACTIONS;

THE OPERATIONS PERSONNEL AND EQUIPMENT THAT ADDRESS THE CURRENT OPERATION WILL NOT CHANGE. NO ADDITIONAL SUPPORT PERSONNEL OR EQUIPMENT ARE REQUIRED TO CARRY OUT THE ACTIONS TO PERFORM THE IA MALFUNCTION PROCEDURE STEPS. THEREFORE, THERE IS NO ADVERSE IMPACT.

(7) A DESCRIPTION OF INFORMATION REQUIRED BY THE CONTROL ROOM STAFF TO DETERMINE WHETHER SUCH OPERATOR ACTION IS REQUIRED, INCLUDING QUALIFIED INSTRUMENTATION USED TO DIAGNOSE THE SITUATION AND TO VERIFY THAT THE REQUIRED ACTION HAS SUCCESSFULLY BEEN TAKEN;

THE CONTROL ROOM STAFF WOULD HAVE INDICATION OF A LOSS OF INSTRUMENT AIR IN ACCORDANCE WITH PROCEDURE OP-901-511, INSTRUMENT AIR MALFUNCTION. RADIATION MONITORING IS ALREADY PROVIDED FOR THE VARIOUS ROOMS TO INDICATE WHETHER THEY ARE ACCESSIBLE. POST RETURN TO SERVICE ACTIONS FOR EC-41355 ARE ESTABLISHED TO TRACK PROCEDURE CHANGES AND TRAINING NECESSARY TO PROTECT THE ASSUMPTIONS IN THE ANALYSIS. THEREFORE, THERE IS NO ADVERSE IMPACT.

(8) THE ABILITY TO RECOVER FROM CREDIBLE ERRORS IN PERFORMANCE OF MANUAL ACTIONS, AND THE EXPECTED TIME REQUIRED TO MAKE SUCH A RECOVERY;

THE ABILITY TO RECOVER FROM CREDITABLE ERROR WILL BE IMMEDIATELY IDENTIFIED BY THE CONTROL ROOM OPERATORS BASED UPON FLOW, TEMPERATURE, AND / OR LEVEL INDICATIONS ASSOCIATED WITH THE VALVE BEING MANIPULATED. FOR EXAMPLE, THE CONTROL ROOM WOULD MONITOR CCW SURGE TANK LEVEL WHILE OPERATORS REALLIGNED SPENT FUEL POOL COOLING AND THE NON-ESSENTIAL CCW LOOP. THE PLACEMENT OF EFW FLOW CONTROL VALVES AND ADVS IS ALREADY A CREDITED MANUAL OPERATOR ACTION SO THIS CHANGE WILL NOT AFFECT THE CREDIBLE ERRORS THAT ALREADY EXISTED. THEREFORE, THERE IS NO ADVERSE IMPACT.

(9) CONSIDERATION OF THE RISK SIGNIFICANCE OF THE PROPOSED OPERATOR ACTIONS;

THE RISK SIGNIFICANCE FOR RESPONDING TO A LOSS OF IA HAS NOT CHANGED AND THE CURRENT LICENSE BASIS ALREADY INCLUDED MANUAL OPERATOR ACTION OUTSIDE OF THE CONTROL ROOM FOR LINING UP ESSENTIAL INSTRUMENT AIR. THIS CHANGE CLARIFIES THOSE ACTIONS. THEREFORE, THERE IS NO ADVERSE IMPACT.

(10) TIME RESPONSE AS OUTLINED IN ANSI/ANS-58.8-1994, "TIME RESPONSE DESIGN CRITERIA FOR SAFETY-RELATED OPERATOR ACTION";

ANSI/ANS-58.8-1994 PROVIDES TIME REQUIREMENTS FOR DIFFERENT PLANT CONDITIONS AND EXPLICITLY LIMITS ITS INTENDED APPLICATION TO TIME CRITICAL OPERATOR ACTIONS. THIS CHANGE ONLY IMPACTS THE OPERATOR ACTIONS THAT WOULD BE REQUIRED TO RESPOND TO A LOSS OF INSTRUMENT AIR. THE TIME TO INITIATE THE ACTIONS TO LINE UP BACKUP AIR SUPPLIES OR POSITION MANUAL HANDWHEEL OVERRIDES IS ON THE ORDER OF EIGHT TO NINE HOURS AFTER THE INDICATION OF THE EVENT. THEREFORE, THE TIME AVAILABLE EXCEEDS THE

MINIMUM TIME REQUIREMENTS CONTAINED IN ANSI 58.8 AND THE OPERATOR ACTION IS CREDIBLE.

EC-41355 / ECS09-005 EVALUTES THAT THE NEW MANUAL ACTION CAN BE ACCOMPLISHED PRIOR TO THE TIME THAT THE SAFETY RELATED MOTIVE GAS ACCUMULATORS ARE EXHAUSTED AND THE AREAS OF THE VARIOUS VALVES ARE FULLY ACCESSIBLE IN ACCIDENT SCENARIOS OF APPLICABLE SAFETY ANALYSES.

CALCULATION ECS09-005 IS THE DESIGN BASIS CALCULATION FOR THE BACKUP AIR SUPPLY MISSION TIME AND DOCUMENTS THE EVALUATION OF THE TIME FOR OPERATOR ACTION. THEREFORE, THERE IS NO ADVERSE IMPACT.

THEREFORE, ENGINEERED AND ADMINISTRATIVE CONTROLS ARE SUFFICIENT TO ENSURE THAT POTENTIAL MALFUNCTIONS ARE NOT INTRODUCED. THE PROPOSED CHANGE ENHANCES EQUIPMENT AND PROCEDURES AND DOES NOT INVOLVE A PHYSICAL PLANT CHANGE. THEREFORE, THE CHANGE OF PLANT OPERATOR ACTION TO EITHER LINE UP BACKUP AIR SUPPLIES OR TO USE HANDWHEEL OVERRIDES FOR CONTROLLING THE SUBJECT VALVES DOES NOT RESULT IN MORE THAN A MINIMAL INCREASE IN THE LIKELIHOOD OF AN 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: EC 41355 CLARIFIES THE MISSION TIME AND SAFETY FUNCTION OF SAFETY RELATED MOTIVE GAS ACCUMULATORS AND EVALUATES APPROPRIATE TESTING REQUIREMENTS OF SEVERAL MANUAL LOCAL HANDWHEEL OVERRIDES ASSOCIATED WITH AIR OPERATED VALVES (AOV) LISTED IN THE DESCRIPTION OF CHANGE SECTION ABOVE, WHICH MAY FAIL OPPOSITE THEIR SAFE POSITION UPON EXHAUSTING THE MOTIVE AIR IN THEIR ASSOCIATED ACCUMULATOR IN A POSTULATED DESIGN BASIS EVENT WITH AN EXTENDED LOSS OF IA. THE CHAPTER 15 ACCIDENTS EVALUATED IN THE UFSAR THAT MAY BE AFFECTED BY THE OPERATION OF THE SUBJECT VALVES ARE LOSS OF COOLANT ACCIDENT (LOCA) (FSAR 15.6.3.3) AND OTHER EVENTS THAT REQUIRE COMPONENT COOLING AND COULD REQUIRE EMERGENCY FEEDWATER AND STEAM GENERATOR COOLDOWN. EACH OF THE SUBJECT VALVES IS NORMALLY REMOTELY OPERATED IN AUTOMATIC USING NORMAL INSTRUMENT AIR AS THE MOTIVE GAS. SAFETY RELATED MOTIVE GAS ACCUMULATORS ARE SIZED TO PROVIDE 10 HOURS OF VALVE OPERATION AFTER A LOSS OF NORMAL INSTRUMENT AIR. THE EXISTING IA MALFUNCTION PROCEDURE REQUIRES ACTIONS TO ENTER TS ACTION STATEMENTS PRIOR TO REACHING THE MINIMUM REQUIRED AIR PRESSURE IN THE ACCUMULATORS THAT ENSURES THE DESIGN BASIS 10 HOUR MISSION TIME. THE EXISTING DESIGN BASIS CREDITS OPERATOR ACTIONS OUTSIDE THE CONTROL ROOM FOR ALIGNING BACKUP AIR SUPPLIES AND USING MANUAL HANDWHEEL OVERRIDES, ALL OF WHICH ARE EVALUATED AS BEING ACCESSIBLE. THE OPERATOR ACTION WILL BE IN RESPONSE TO LOSS OF INSTRUMENT AIR THAT HAS AN EXISTING PLANT ANALYSIS RESPONSE THAT REMAINS UNCHANGED. THE ACTION OF EITHER ALIGNING BACKUP AIR SUPPLIES OR USING MANUAL HANDWHEEL OVERRIDES WILL BE REFLECTED IN OPERATIONS PROCEDURES AND OPERATOR TRAINING PROGRAMS. THE MANUAL ACTION CAN BE ACCOMPLISHED PRIOR THE 10 HOUR TIME CAPACITY OF THE SAFETY RELATED MOTIVE GAS ACCUMULATORS. A CREDIBLE ERROR IS UNLIKELY SINCE IN MOST CASES THE ACTION IS SIMPLY TO GAG A VALVE IN ITS SAFE POSITION OR LINE UP A BACKUP AIR SUPPLY TO THE SAFETY RELATED MOTIVE GAS ACCUMULATORS. IN THE CASE OF THE EMERGENCY FEEDWATER (EFW) CONTROL VALVES OR ATMOSPHERIC DUMP VALVES (ADV), OPERATORS WILL BE IN CONSTANT COMMUNICATION WITH THE CONTROL ROOM VIA RADIOS AND REACTING TO CONTROL ROOM INSTRUCTIONS TO OPEN AND CLOSE THE VALVES TO CONTROL REACTOR COOLANT SYSTEM (RCS) TEMPERATURE AND STEAM GENERATOR LEVEL. THE PROPOSED CHANGE ENHANCES THE POST-LOSS OF INSTRUMENT AIR PERFORMANCE OF SAFETY RELATED VALVES. 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: FSAR TABLE 9.2-4 LISTS THE FAILURE MODES AND EFFECTS ANALYSIS (FMEA) FOR THE CCW AND ACCW SYSTEMS. THE FMEA POSTULATES LOSS OF AIR. CURRENTLY THE FMEA STATES THAT "THE ACTIVE VALVES ARE PROVIDED WITH ACCUMULATORS AND / OR SPRINGS TO ENSURE PROPER ALIGNMENT OF THE SYSTEM." THE CONSEQUENSE OF A MALFUNCTION WHERE THE CROSS-CONNECT VALVES FAIL OPEN PRIOR TO OPERATOR TAKING ACTION TO GAG THE VALVE CLOSED COULD POSSIBLY RESULT IN DISABLING OF THE OPERATING TRAIN OF ULTIMATE HEAT SINK. HOWEVER, AS DISCUSSED IN QUESTION 2 AND QUESTION 5, THE OPERATOR ACTION IS ACHEIVABLE. THEREFORE, THE FOLLOWING STATEMENT WILL BE ADDED: "SAFETY RELATED ACCUMULATORS ARE

CAPABLE OF PROVIDING MOTIVE AIR TO PNEUMATICALLY OPERATED VALVES FOR 10 HOURS. PROCEDURES ARE ESTABLISHED FOR OPERATING MANUAL HANDWHEEL OVERRIDES OR LINING UP BACKUP AIR SUPPLIES FOR CONTINUED SAFETY FUNCTION AFTER 10 HOURS."

FSAR TABLE 10.4-14 LISTS THE FMEA FOR THE EFW SYSTEM. THE FMEA POSTULATES FAILURE OF AN EFS VALVE TO OPERATE. THE FMEA CONCLUDES THAT THERE IS NO EFFECT BECAUSE THE REDUNDANT VALVES FLOW PATHS PERMIT OPERATION OF THE SYSTEM. THIS CHANGE SUPPORTS THAT CONCLUSION BECAUSE THE MANUAL HANDWHEEL OVERRIDE WILL BE CREDITED ON BOTH EFW FLOW CONTROL VALVES ON BOTH TRAINS.

FSAR 9.3.6.3.3 DISCUSSES THE AFFECT OF LOSS OF IA ON AN ADV AND THE EFFECT ON ABILITY TO ACHIEVE SHUTDOWN COOLING (SDC) ENTRY CONDITIONS. THE CURRENT ANALYSIS CREDITS THE ABILITY TO INITIATE SDC WHEN RCS TEMPERATURE OF 400°F, WHICH WILL OCCUR BEFORE THE 10 HOUR BACKUP SUPPLY OF MOTIVE GAS FOR THE ADV IS EXHAUSTED FOR THE CASE OF A SINGLE AVAILABLE ADV. THE FSAR ALSO CURRENTLY CLARIFIES THAT IT IS DESIRABLE TO ACHIEVE ≤350°F PRIOR TO INITIATING SDC. THEREFORE, THE FOLLOWING STATEMENT WILL BE ADDED: "PROCEDURES ARE ESTABLISHED FOR OPERATING MANUAL HANDWHEEL OVERRIDES OR LINING UP BACKUP AIR SUPPLIES FOR CONTINUED SAFETY FUNCTION AFTER 10 HOURS."

FSAR TABLE 6.3-1 LISTS THE FMEA FOR THE SAFETY INJECTION (SI) SYSTEM. THE FMEA POSTULATES FAILURE OF THE ADV TO OPEN DUE TO CONTROL LOOP FAILURE, WHICH WOULD INCLUDE LOSS OF AIR. THE FMEA CONCLUDES THAT THE RUSULT IS NO EFFECT BECAUSE THE OTHER ADV IS AVAILABLE. THIS CHANGE SUPPORTS THAT CONCLUSION BECAUSE THE MANUAL HANDWHEEL OVERRIDE WILL BE CREDITED ON BOTH ADVS.

THE INHERENT COMPENSATING PROVISION REMAINS THE REDUNDANT TRAIN WITH THE PROPOSED CHANGE. OPERATOR ACTION IS CREDITED TO EITHER LINE UP BACKUP AIR SUPPLIES OR OPERATE HANDWHEEL OVERRIDES TO ALLOW OPERATION OF VALVES AFTER THE 10 HOUR MOTIVE GAS SUPPLY IS EXHAUSTED. IN EACH CASE THE DESIGN MAINTAINS QUALIFICATION, SEPARATION, AND REDUNDANCY OF EQUIPMENT. 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: THE CHAPTER 15 ACCIDENTS EVALUATED IN THE UFSAR THAT MAY BE AFFECTED BY THE OPERATION OF THE SUBJECT VALVES ARE LOSS OF COOLANT ACCIDENT (LOCA) (FSAR 15.6.3.3), AND OTHER EVENTS THAT REQUIRE COMPONENT COOLING AND COULD REQUIRE EMERGENCY FEEDWATER AND STEAM GENERATOR COOLDOWN. THE POTENTIAL MALFUNCTIONS INTRODUCED BY CREDITING OPERATOR ACTION AFTER THE 10 HOUR MOTIVE GAS ACCUMULATORS ARE EXHAUSTED ARE 1) MISPOSITIONING OF THE SUBJECT VALVE(S), AND 2) FAILURE TO TAKE ACTION TO MAINTAIN CONTROL OF THE VALVE POSITION AFTER THE 10 HOUR MOTIVE GAS ACCUMULATORS ARE EXHAUSTED. EC 41355 CLARIFIES THE MISSION TIME AND SAFETY FUNCTION OF SAFETY RELATED MOTIVE GAS ACCUMULATORS AND EVALUATES APPROPRIATE TESTING REQUIREMENTS OF SEVERAL MANUAL LOCAL HANDWHEEL OVERRIDES ASSOCIATED WITH AIR OPERATED VALVES (AOV) LISTED IN THE DESCRIPTION OF CHANGE SECTION ABOVE, WHICH MAY FAIL OPPOSITE THEIR SAFE POSITION UPON EXHAUSTING THE MOTIVE AIR IN THEIR ASSOCIATED ACCUMULATOR IN A POSTULATED DESIGN BASIS EVENT WITH AN EXTENDED LOSS OF IA. ENGINEERED AND ADMINISTRATIVE CONTROLS ARE DESIGNED TO ENSURE THAT POTENTIAL MALFUNCTIONS ARE NOT INTRODUCED. POSITIONING ANY OF THE SUBJECT VALVES IS NOT AN ACCIDENT INITIATOR. THE PROPOSED CHANGE DOES NOT INTRODUCE ANY FACTORS MAKING EXISTING TRANSIENTS AND ACCIDENTS MORE LIKELY IN THAT IT MAINTAINS SEPARATE AND INDEPENDENT CCW, ACCW, EFW, AND ADV TRAINS AND CONTROL OF THE SUBJECT VALVES IS BY OPERATORS TRAINED IN THE APPROVED PROCEDURES.

THEREFORE, THE PROPOSED CHANGE DOES NOT 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: THIS CHANGE AFFECTS AOVS THAT FAIL OPPOSITE THEIR SAFE POSITION. MANUAL OPERATOR ACTION OUTSIDE OF THE CONTROL ROOM IS CREDITED TO MAINTAIN CONTROL OF THE VALVE POSITION FOLLOWING A LOSS

OF IA AND EXHAUSTING THE 10 HOUR MOTIVE GAS ACCUMULATOR. THE INHERENT COMPENSATING PROVISION REMAINS THE REDUNDANT AND SEPARATE EQUIPMENT TRAINS WITH THE PROPOSED CHANGE. THEREFORE THE EXISTING FMEA TABLES AND DISCUSSION REGARDING THE EFFECTS OF A LOSS OF IA SYSTEM REMAIN VALID WITH CLARIFICATION REGARDING THE MANUAL OPERATOR ACTIONS.

THEREFORE, THE PROPOSED CHANGE 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 DESIGN BASIS LIMITS FOR FISSION PRODUCT BARRIERS ARE NOT IMPACTED. EC 41355 CLARIFIES THE MISSION TIME AND SAFETY FUNCTION OF SAFETY RELATED MOTIVE GAS ACCUMULATORS AND EVALUATES APPROPRIATE TESTING REQUIREMENTS OF SEVERAL MANUAL LOCAL HANDWHEEL OVERRIDES ASSOCIATED WITH AIR OPERATED VALVES (AOV) LISTED IN THE DESCRIPTION OF CHANGE SECTION ABOVE, WHICH MAY FAIL OPPOSITE THEIR SAFE POSITION UPON EXHAUSTING THE MOTIVE AIR IN THEIR ASSOCIATED ACCUMULATOR IN A POSTULATED DESIGN BASIS EVENT WITH AN EXTENDED LOSS OF IA. OPENING OR CLOSING ANY OF THE SUBJECT VALVE IS NOT AN ACCIDENT INITIATOR. THE FMEA DESCRIBED IN THE FSAR REMAIN THE SAME WITH CLARIFICATION REGARDING THE MANUAL OPERATOR ACTIONS. MANUAL OPERATOR ACTION HAS BEEN EVALUATED AS ACHIEVABLE. THE INHERENT COMPENSATING PROVISION REMAINS THE REDUNDANT TRAIN WITH THE PROPOSED CHANGE. OPERATOR ACTION IS CREDITED TO EITHER LINE UP BACKUP AIR SUPPLIES OR OPERATE HANDWHEEL OVERRIDES TO ALLOW OPERATION OF VALVES AFTER THE 10 HOUR MOTIVE GAS SUPPLY IS EXHAUSTED. IN EACH CASE THE DESIGN MAINTAINS QUALIFICATION, SEPARATION, AND REDUNDANCY OF EQUIPMENT. THEREFORE, THIS CHANGE HAS NO IMPACT ON ANY OF THE DESIGN BASIS LIMITS FOR FISSION PRODUCT BARRIERS AS THE CHANGE IS TO MITIGATIVE SYSTEMS TO ENHANCE PERFORMANCE DURING LOSS OF INSTRUMENT AIR IN ACCIDENT RECOVERY. THEREFORE, THERE IS NO IMPACT ON THE DESIGN BASIS LIMITS FOR THE FUEL CLADDING, THE REACTOR COOLANT SYSTEM PRESSURE BOUNDARY, OR THE CONTAINMENT. THEREFORE, THIS CHANGE DOES NOT IMPACT ANY LIMIT FOR A FISSION PRODUCT BARRIER AS DESCRIBED IN THE UFSAR.

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 CALCULATION METHODS USED FOR EVALUATING THE ACCESSIBILITY OF THE VALVES AND ACCUMULATORS, INCLUDING THE DOSE RATES IN THE AREA ARE NOT CHANGED. 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.

Evaluation # 2013-01

I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # 2013-01 / Rev. #:** 0

Proposed Change / Document: EC-43821, Provide Regulated N2 Supply to SIT 2B

Description of Change: Install a temporary alternate supply of nitrogen to Safety Injection Tank 2B (SIT 2B). The Temporary modification will utilize the Nitrogen supply header, which enters the Containment through Containment Isolation valves NG-157 and NG-158 (inside containment isolation check valve), and installing a non-nuclear safety related connecting piping / hose and regulator, and two (2) safety related check valves in series (to preclude depressurizing SIT 2B following a failure of the NNS portion of the nitrogen supply), which will be connected to the safety related instrument tubing for SI IPT0343, "SI tank 2B Pressure Transmitter", through the instrument line at SI IPT343. SI IPT343 is the Narrow Range pressure transmitter for SIT 2B. Note that the Temp. Mod. disables SI IPT343 for pressure sensing purposes. SI IPT343 (& associated PMC pt ID A44405) is utilized for Tech Spec compliance IAW OP-903-001. However, Pressure Transmitter SI IPT342 (SI Tank 2B Pressure Transmitter) provides redundant indication to SI IPT343.

The Temporary Modification is being installed to supply a continuous makeup of nitrogen to SIT 2B to makeup for existing (N2) leakage from SIT 2B, which is causing SIT 2B to depressurize, and requires operator action to makeup N2 to SIT2B several times per day (reference CR-WF3-2013-1210). The bulk nitrogen supply system is not normally used during normal plant operation as a continuous nitrogen supply. The design of the Nitrogen supply system is to be used on as needed basis via remote manual operation.

Summary of Evaluation: Evaluation has concluded that EC 43821 will not adversely affect the design or function of the Safety Injection System.

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: Peter McKenna / /EOI/System Engineering -NSSS/ 4 - 18 - 2013
Name (print) / Signature / Company / Department / Date

Reviewer: Ken Boudreaux/ /EOI/System Engineering -NSSS/ 4 - 18 - 2013
Name (print) / Signature / Company / Department / Date

OSRC: Carl Rich/ /EOI/Nuclear Safety Assurance Director/ 4 -19 - 2013
Chairman's Name (print) / Signature / Date

W3-13-10
OSRC Meeting #

[1] 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.

Evaluation # 2013-01

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: SIT 2B IS NOT AN ACCIDENT INITIATOR. THE TEMP. MOD. IS NOT CHANGING SIT 2B, NOR ANY OTHER SSC SUCH THAT IT WILL ALTER WHETHER IT WILL BECOME AN ACCIDENT INITIATOR. ALSO THE CONTAINMENT ISOLATION AND NITROGEN BULK GAS (NG) SYSTEMS ARE NOT ACCIDENT INITIATORS. THE TEMP MOD WILL CHANGE THE NITROGEN SUPPLY TO SIT 2B SUCH THAT IT WILL SUPPLY A CONTINUOUS NITROGEN FEED TO SIT 2B, DURING NORMAL PLANT OPERATION, AND IN THE EVENT THAT NG-157 (CONTAINMENT N2 SUPPLY OUTSIDE ISOLATION) IS NOT CLOSED, OR FAILS TO CLOSE UPON RECEIPOT OF A CIAS.

THE TEMPORARY MODIFICATION PER EC-43821 WILL INSTALL PIPING AND COMPONENTS TO ROUTE NITROGEN FROM THE NITROGEN SUPPLY HEADER AT VALVE NG-502 (CONTAINMENT 750/50 PSIG PCV INLET DRAIN), TO SIT 2B AT INSTUMENT SI IPT343 ASSOCIATED TUBING, FOR A CONTINUOUS NITROGEN SUPPLY TO SIT 2B. A REVIEW WAS CONDUCTED FOR A SINGLE FAILURE OF THE NITROGEN SUPPLY SYSTEM COMPONENTS ALONG WITH SIT 2B COMPONENTS TO DETERMINE IF THE FREQUENCY OF OCCURRENCE OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR MAY BE INCREASED. EC 43821 WILL INSTALL A TEMPORARY ALTERNATE SOURCE OF NITROGEN TO SAFETY INJECTION TANK (SIT) 2B IN ORDER TO MAINTAIN TECHNICAL SPECIFICATION REQUIRED NITROGEN COVER GAS PRESSURE IAW TS 3.5.1. THE SAFETY INJECTION TANKS (SITS) ARE REQUIRED TO MAINTAIN PRESSURE BETWEEN 600 AND 670 PSIG TO ENSURE ADEQUATE PRESSURE TO FLOOD THE CORE IN THE EVENT OF A LOSS OF COOLANT ACCIDENT (LOCA). THIS ALTERNATE NITROGEN SOURCE WILL BE REGULATED BELOW THE MAXIMUM PRESSURE (670 PSIG) SPECIFIED BY TECHNICAL SPECIFICATION 3/4.5.1 FOR SIT 2B. THE TANK HAS A MAXIMUM DESIGN PRESSURE RATING OF 700 PSIG AND IS SUPPLIED WITH A RELIEF VALVE SET AT 700 PSIG WITH A DESIGN FLOWRATE OF 1120 SCFM (PER UFSAR 6.3.2.2.1), HOWEVER, PER CALCULATION ECM98-006, THE MAXIMUM DISCHARGE FLOW CAPABILITY OF RELIEF VALVE SI-328B (SIT 2B RELIEF TO CONTAINMENT SUMP) AT ITS SET PRESSURE IS 6429.98 SCFM. THE MAXIMUM AVAILABLE FLOWRATE FROM THE ALTERNATE SOURCE (CONTINUOUS N2 SUPPLY) IS 150 SCFM (REFERENCE EC-43821 FOR THE MAXIMUM CONTINUOUS N2 FLOW RATE). THE NITROGEN SUPPLY SYSTEM IS PROVIDED WITH A SAFETY CLASS 2, FULL CAPACITY SAFETY RELIEF; NG-1523 (750 PSIG N2 SUPPLY HEADER RELIEF), TO ENSURE THAT, IN THE EVENT OF A FAILURE OF THE NON-SAFETY RELATED PRESSURE REGULATOR(S), NG-147A(B) (NITROGEN MANIFOLD REGULATOR A & B), AND RELIEF VALVE NG-149 (N2 MANIFOLD OUTLET RELIEF), THAT THE NITROGEN SUPPLY CANNOT OVERPRESSURIZE ITS SERVICED COMPONENTS DURING NITROGEN FILL EVOLUTION(S). IN THE EVENT OF A FAILURE OF NG-1523, SIT RELIEF VALVE SI-328B WILL PROTECT SIT 2B FROM NITROGEN GAS OVERPRESSURIZATION SINCE THE CAPACITY OF SI-328B ENVELOPES THE MAXIMUM N2 FLOW RATE WHICH CAN OCCUR IN THE EVENT OF A FAILURE OF UPSTREAM PRESSURE REGULATING DEVICES AND/OR RELIEF VALVES. PER UFSAR 6.3.2.5.4 "RELIEF VALVE ... FAILURES ARE NOT CONSIDERED CREDIBLE FAILURES". THE SIT IS ISOLATED FROM THE RCS AT CHECK VALVE SI-330B (SIT 2B OUTLET CHECK). THE TEMPORARY MODIFICATION DOES NOT CHANGE OR ALTER CONDITIONS AT SI-330B, WHICH WOULD BE PRESENT DURING ANY NITROGEN MAKEUP EVOLUTION.

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THEREFORE PRESSURE IN THE TANK WILL BE MAINTAINED LOWER THAN ITS MAXIMUM DESIGN PRESSURE. NITROGEN GAS (NG) RELIEF VALVE NG-1523 IS PROVIDED IN THE UPSTREAM N2 SUPPLY HEADER TO PROTECT THE NITROGEN SYSTEM SUPPORTED COMPONENTS FROM OVERPRESSURIZATION (INDUS PASSPORT VALIDATED PARAMETERS). TWO SAFETY RELATED CHECK VALVES, INSTALLED IN SERIES, IN THE N2 SUPPLY, WILL PROTECT SIT 2B, THE PRESSURE TRANSMITTERS, AND THE NITROGEN SUPPLY PIPING INSTALLED PER EC-43821 FROM A PASSIVE FAILURE OF THE CONNECTING NON-SAFETY RELATED PIPING AND (ASSUMING A SINGLE ACTIVE FAILURE OF ONE CHECK VALVE) FROM DEPRESSURIZING.

THE SAFETY RELATED ADDITIONS (E.G. CHECK VALVES) MADE BY EC-43821 ARE EVALUATED TO ENSURE THAT THE TEMPORARY MODIFICATION DOES NOT ADVERSELY AFFECT THE EXISTING, AS WELL AS TEMPORARY, SAFETY RELATED SSC (SEE EC-43821) FOR SEISMIC AND PIPE STRESS AFFECTS. THEREFORE THERE IS NO INCREASE IN THE FREQUENCY OF OCCURRENCE OF A DESIGN BASIS ACCIDENT (DBA) AS A RESULT OF THIS TEMP MOD.

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: NARROW RANGE INSTRUMENTS SI-IPT0342 (SI TANK 2B PRESSURE TRANSMITTER) AND SI IPT0343 ARE NOT CREDITED FOR POST ACCIDENT MONITORING OF SIT 2B PRESSURE PER UFSAR TABLE 7.5-3 "ACCIDENT MONITORING INSTRUMENTATION". THE TEMPORARY MODIFICATION PER EC-43821 WILL DISABLE NARROW RANGE TRANSMITTER SI IPT0343. THIS TRANSMITTER IS DESCRIBED IN THE UFSAR 5.2.5.1.4, WHICH IDENTIFIES THAT THE SAFETY INJECTION TANK PRESSURES ARE MONITORED BY THREE (3) PRESSURE TRANSMITTERS, INCLUDING SI IPT0343, FOR SIT 2B, IN ORDER TO PROVIDE DETECTION OF REACTOR COOLANT (RCS) INLEAKAGE, PRESSURE REQUIRED PER TS 3/4.5.1, & (POST ACCIDENT - FOR SI IPT341 [SI TANK 2B PRESSURE TRANSMITTER]) POST ACCIDENT SIT PRESSURE. TEMPORARY MODIFICATION EC-43821 IS DISABLING SI-IPT0343, IN ORDER TO UTILIZE THE CORRESPONDING INSTRUMENT TUBING TO PROVIDE THE CONTINUOUS NITROGEN FEED. THE ABILITY TO MONITOR SIT 2B PRESSURE WILL BE PROVIDED VIA THE REMAINING NARROW RANGE TRANSMITTER, SI IPT342, AND THE DISABLING OF SI IPT343 DOES NOT DISABLE A FUNCTION PREVIOUSLY EVALUATED IN THE UFSAR.

THE PIPING/TUBING ADDED BY THE TEMPORARY MODIFICATION INCLUDES TWO SAFETY RELATED CHECK VALVES, AND CONNECTING PIPING/TUBING TO SI IPT343, WHICH WILL BE DISABLED FOR PRESSURE MONITORING. PRESSURE TRANSMITTER SI IPT342 WILL BE CONNECTED TO WIDE RANGE TRANSMITTER SI IPT341 VIA A SAFETY RELATED JUMPER. THE SAFETY RELATED BOUNDARY WILL BE AT THE TWO CHECK VALVES WHICH PROVIDE REDUNDANT ISOLATION FROM AN EGRESS EVENT DUE TO A FAILURE OF UPSTREAM NON-SAFETY RELATED PIPING OR TUBING.

THE TEMPORARY MODIFICATION WILL NOT ADVERSELY AFFECT THE REMAINING PRESSURE TRANSMITTERS, SI IPT0342 (NARROW RANGE) AND SI IPT0341 (WIDE RANGE), AND CAN THEREFORE PROVIDE INDICATION OF SIT 2B PRESSURE TO MEET TS 3/4.5.1, TO DETECT RCS INGRESS (BACKLEAKAGE) INTO THE SIT 2B & ITS' ASSOCIATED PIPING, TO DETECT AND ENSURE REDUCED SIT 2B PRESSURE, TO PREVENT OVERPRESSURIZATION OF THE SHUTDOWN COOLING (SDC) SYSTEM AS DESCRIBED IN UFSAR 6.3.2.2.1, AND TECHNICAL SPECIFICATION 3/4.5.1. SI IPT343 PROVIDES HI-HI AND LO-LO ALARMS, WHICH WILL DISABLED PER EC-43821.

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THE REMAINING NARROW RANGE INSTRUMENT, SI IPT342, RETAINS THE ABILITY TO MONITOR SIT PRESSURE TO ENSURE THE MAXIMUM SIT PRESSURE DOES NOT EXCEED 670 PSIG. OP-903-001 ENSURES SHIFTLY CHECKS ARE MADE; LO AND HI PRESSURE ALARMS WILL ANNUNCIATE IN THE EVENT THAT PRESSURE FALLS OUTSIDE THE NORMAL OPERATING PRESSURE BAND. PER TS 3/4.5.1 WITH PRESSURIZER PRESSURE LESS THAN 1750 PSIA THE MINIMUM SIT PRESSURE WILL BE 235 PSIG, AND PER UFSAR 6.3.2.2.1, WHEN ENTERING SDC OPERATION, THE SIT PRESSURE WILL BE LOWERED TO BETWEEN 235 AND 300 PSIG. AS PER THE UFSAR, TABLE 7.5-3, THE POST ACCIDENT WIDE RANGE PRESSURE INDICATION PROVIDED BY SI IPT0341 IS NOT ADVERSELY AFFECTED BY THE TEMPORARY MODIFICATION AND WILL PROVIDE THE REQUIRED INDICATION FOR REDUCED PRESSURE OPERATION AND POST ACCIDENT INDICATION AS PER UFSAR TABLE 7.5-3.

IN ADDITION, THE EC-43821 N2 ADDITION FLOW PATH WILL NOT BE INTERCONNECTED WITH NARROW RANGE TRANSMITTER SI IPT0342 TO MITIGATE PRESSURE FLUCTUATIONS OR SPURIOUS PRESSURE READINGS DUE TO FLOW EFFECTS, AND/OR N2 MIGRATION INTO THE TRASMITTER REFERENCE LEG FROM THE TEMPORARY MODIFICATION. THE PRESSURE TRANSMITTERS CONNECT TO SIT 2B AT THE 1" DIAMETER INLET, WHICH WILL BE ADJACENT TO SIT 2B & RESULT IN NEGLIGIBLE FLOW EFFECTS ON THE PRESSURE READINGS. A SAFETY RELATED TEMPORARY JUMPER WILL CONNECT THE WIDE RANGE TRANSMITTER, SI IPT0341, WITH NARROW RANGE TRANSMITTER SI IPT0342, IN ORDER TO ISOLATE NR TRANSMITTER SI IPT0342 FROM THE N2 FLOWPATH (SEE EC-43821).

DURING AND FOLLOWING A DBA, THE NITROGEN SUPPLY HEADER WAS EXPECTED TO BE OUT OF SERVICE, SINCE THE SYSTEM DESIGN BASIS WAS TO PROVIDE MAKEUP TO THE N2 ACCUMULATORS, AND NOT PROVIDE A CONTINUOUS N2 SUPPLY, THEREFORE THE PROTECTION FOR THE CONTAINMENT PENETRATION ASSOCIATED WITH VALVE NG-157 (CB MPEN-014) WAS/IS PROVIDED BY NORMALLY OPEN (OP-003-019 AND UFSAR 3.9), FAILED CLOSED, SC-2 ISOLATION VALVE NG-157 AND SC-2 CHECK VALVE NG-158 (CONTAINMENT N2 SUPPLY CHECK). THESE VALVES PROVIDE REDUNDANT PROTECTION AGAINST A REACTOR CONTAINMENT EGRESS DURING AND FOLLOWING A DBA.

NOTE THAT THE RESPONSE TO QUESTION 2 FOR AN ADDITIONAL OPERATOR ACTION, TO CLOSE NG-157, IS EVALUATED IN ATTACHMENT 1, FOR FEASIBILITY (REF. ANS 58.8).

TWO HOURS AFTER THE ACCIDENT, IF NOT ALREADY CLOSED VIA RECEIPT OF A CIAS, VALVE NG-157 WILL BE REMOTE MANUALLY ISOLATED (REFERENCE OI-38, 5.2.5, WHICH IMPLEMENTS EMERGENCY OPERATING PROCEDURES [EOPS]). NORMALLY OPEN VALVE NG-157 RECEIVES A CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS) TO CLOSE. SINCE THE TEMPORARY MODIFICATION PROVIDES A CONTINUOUS N2 SUPPLY TO SIT 2B THROUGH NG-157, A SINGLE ACTIVE FAILURE OF NG-157 TO CLOSE, EITHER BY CIAS, OR BY REMOTE MAUNAL ACTUATION, WOULD FACILITATE THE CONTINUOUS FEED OF N2, DURING AND FOLLOWING A WORST CASE DBA. THE LIKELIHOOD OF THE FAILURE OF NG-157 TO CLOSE IS NOT INCREASED AS A RESULT OF THIS MODIFICATION, AND IN THE EVENT OF A LOSS OF N2 HEADER PRESSURE, THE REDUNDANT ISOLATION PROVIDED BY SC-2 CHECK VALVE NG-158, WOULD ENSURE THE CONTAINMENT ISOLATION FUNCTION, AS DESCRIBED IN THE UFSAR TABLE 6.2-32. REFERENCE G166 SHEET 1.

THE OPERATOR ACTION TO SECURE THE CONTINUOUS NITROGEN FEED WITHIN TWO HOURS FOLLOWING A DESIGN BASIS ACCIDENT (DBA) WILL ENSURE THAT EXCESS NITROGEN SATURATED WATER DOES NOT ACCUMULATE IN THE RCS AND/OR THE SAFETY INJECTION SUMP. THIS OPERATOR ACTION ALSO ENSURES OVERPRESSURE PROTECTION FOR THE SDC IS PROVIDED, ALTHOUGH EXISTING PROCEDURAL CONTROLS EXIST TO PRECLUDE THE SIT PRESSURE FROM CAUSING A SDC OVERPRESSURE EVENT (VIA ISOLATION OF THE SITS FROM THE RCS).

THEREFORE THERE WILL BE NO INCREASE IN THE LIKELIHOOD OF OCCURRENCE OF A MALFUNCTION OF AN SSC IMPORTANT TO SAFETY AS A RESULT OF THE TEMPORARY MODIFICATION PER EC 43821. EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE, POST DESIGN BASIS ACCIDENT (DBA), WITH RESPECT TO NITROGEN INFILTRATION; THE IMPACT UPON THE ECCS UPON A DESIGN BASIS ACCIDENT (LOSS OF COOLANT ACCIDENT - LOCA) IS PROVIDED IN THE RESPONSE TO QUESTIONS 3, 6 AND 7, BELOW.

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THE PRESENCE OF THE ADDITIONAL NITROGEN GAS DUE TO THE CONTINUOUS N2 FEED WILL BE EVALUATED FOR THE EFFECT UPON ECCS SYSTEM PERFORMANCE AND LONG TERM HEAT REMOVAL: FOLLOWING A DESIGN BASIS ACCIDENT, IT IS ASSUMED THAT A LARGE FRACTION OF THE SIT COVER GAS INGRESSES INTO THE RCS. IN ADDITION OPERATING EXPERIENCE INDICATES THAT THE SIT WATER MAY BE FULLY SATURATED WITH NITROGEN DUE TO THE PROXIMITY OF THE SIT INVENTORY TO THE N2 COVER GAS AND THE N2 COVER GAS PRESSURE. WESTINGHOUSE LETTER LTR-LIS-08-543 PREVIOUSLY EVALUATED THE EFFECT OF LARGE GAS VOIDS ON THE CORE AND POST ACCIDENT RECOVERY UPON CORE THERMAL-HYDRAULIC PERFORMANCE AS A RESULT OF THE EXISTING N2 COVER GAS IN THE SITS, AND CONCLUDED THAT NO ADVERSE EFFECTS WILL RESULT. THE ADDITIONAL GAS ADDED DURING A (2) HOUR PERIOD, BEFORE OPERATOR ACTION WOULD ISOLATE THE N2 HEADER AT NG-151 (RAB 750 PSIG N2 SUPPLY ISOLATION) OR NG-152 (RAB 750 PSIG N2 SUPPLY) (SEE ATTACHMENT 1 TO THIS EVALUATION), WITH SIT 2B RECEIVING A CONTINUOUS FEED OF N2 VIA AN OPEN (ASSUMING A SINGLE ACTIVE FAILURE TO CLOSE) NG-157 WOULD RESULT IN AN ADDITIONAL 18,000 SCF (150 SCFM MAXIMUM X 120 MINUTES).

WESTINGHOUSE LETTER CWTR3-13-26 EVALUATES THE EFFECT OF THE CONTINUOUS N2 GAS SUPPLY ON CORE COOLING DURING DBA. THE IMPACT UPON CORE COOLING AND THE DESIGN BASIS ACCIDENT [DBA] IS DISCUSSED IN THE RESPONSE TO QUESTIONS 3, 6, AND 7, BELOW). DURING THE RECIRCULATION PHASE OF THE DESIGN BASIS ACCIDENT (DBA), WHICH WOULD OCCUR WHEN THE REFUELING WATER STORAGE POOL (RWSP) INVENTORY REACHES ITS MINIMUM LEVEL, WHICH IS EXPECTED TO OCCUR ~20 MINUTES OR MORE INTO A WORST CASE DBA (LBLOCA), WATER FROM THE SITS, MIXED WITH RCS INVENTORY, AND RWSP INVENTORY, WILL POOL AT THE BOTTOM OF CONTAINMENT, AND WILL PROVIDE A SUCTION SOURCE THROUGH VALVES SI-602A(B), UPON RECEIPT OF AN RECIRCULATION ACTUATION SIGNAL (RAS), TO THE HIGH PRESSURE SAFETY INJECTION (HPSI) SYSTEM PUMPS' AND THE CONTAINMENT SPRAY (CS) PUMPS' SUCTIONS. THE RATIO OF THE POTENTIALLY SATURATED SIT INVENTORY TO OTHER WATER INVENTORY, SUCH AS THE RWSP, IS EXPECTED TO BE GREATER THAN ~ 8:1 (BASED ON A MINIMUM 8:1 RATIO OF RWSP, LPSI, AND RCS TO THE SIT INVENTORY; REFERENCE W3-DBD-001, SAFETY INJECTION SYSTEM DESIGN BASIS DOCUMENT). SINCE THE RWSP INVENTORY DOES NOT INTERFACE WITH PRESSURIZED GAS (THE RWSP IS VENTED TO CHARCOAL FILTERS VIA THE AIR EVACUATION SYSTEM (AES) PER G163 AND G153 SHEET 1); THEREFORE THE RWSP INVENTORY, THE LOW PRESSURE SAFETY INJECTION (LPSI) PIPING INVENTORY, AND THE RCS, IS NOT EXPECTED TO HAVE ANY SIGNIFICANT GAS SATURATION LOAD PRIOR TO, OR DURING, THE DBA. SOME MIXING OF THE SIT INVENTORY, AND POST INJECTION N2 GAS ADDITION, WITH RWSP AND OTHER NON GAS SATURATED INVENTORY, WILL OCCUR, DURING THE RCS INJECTION PHASE, HOWEVER, AN OPERATOR ACTION TO ISOLATE THE NITROGEN BULK GAS SUPPLY HEADER AT NG-151 OR NG-152, WILL BE PERFORMED (2) HOURS AFTER THE DBA, WHICH WILL PRECLUDE N2 GAS FROM REACHING THE SATURATION LIMIT FOR THE TOTAL RECIRCULATION VOLUME DURING RAS. THE PRESSURE DROP BETWEEN THE SI SUMP INLET AND THE PUMP SUCTION EYES (REFERENCE NPSH CALCULATION ECM07-001) WILL BE LESS THAN 8 FEET (< 3.5 PSID), WITH THE SUCTION STRAINER LOSSES EQUALLING LESS THAN 1 FOOT OF HEAD. BASED UPON THE RESIDENCE TIME IN THE SI / CONTAINMENT SUMP BEFORE RAS AND THE EXFILTRATION THAT THE SIT INVENTORY WOULD EXPERIENCE, THE LOW RATIO OF SIT INVENTORY TO RCS AND RWSP INVENTORY, AND THE LOW PRESSURE DROP BETWEEN THE CONTAINMENT AND THE HPSI AND CS PUMP SUCTIONS WOULD RESULT IN NO, OR NONSIGNIFICANT, GAS EXFILTRATION IN THE ECCS DURING RAS OPERATION, AND/OR SDC OPERATION DURING LONG TERM COOLING.

ANY NITROGEN EXFILTRATION THAT COULD OCCUR AT THE HPSI AND CS PUMPS SUCTIONS WOULD THEREFORE BE INSIGNIFICANT AND < 5% OF THE TOTAL ECCS (INCLUDING CS) FLOW. PER THE PUMP HANDBOOK (2ND EDITION, MCGRAW HILL - 1986) PROVIDED AIR ACCUMULATION (VOIDING) IN A FLUID SYSTEM IS LESS THAN 5% OF THE TOTAL FLOW THERE WILL BE NO ADVERSE EFFECT ON CENTRIFUGAL PUMP OPERATION. THE LOWEST PRESSURE POINTS IN THE HPSI OR CS SYSTEMS WILL BE EITHER AT THE SUCTION STRAINER, THE PUMP SUCTION EYE, OR AT THE POINT OF DISCHARGE. SINCE DISCHARGE LOCATIONS WOULD EXFILTRATE TO THE CONTAINMENT ATMOSPHERE, EITHER VIA THE CS SYSTEM SPRAY NOZZLES OR THE RCS BREAK, AND INSUFFICIENT PRESSURE DROP EXISTS TO FACILITATE EXFILTRATION OF NITROGEN INTO THE PUMP SUCTION FLOW STREAM, PER CALCULATION ECM07-001,

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THE PRESENCE OF N2 SATURATED WATER FROM THE SITS WILL NOT ADVERSELY AFFECT THE POST ACCIDENT ECCS PERFORMANCE. FOLLOWING THE DISCHARGE OF SIT 2B INTO THE RCS THE SIT GAS PRESSURE AND RCS PRESSURES WILL REACH AN EQUILIBRIUM CONDITION (EQUILIBRIUM PRESSURE). FOR THE CONTINUOUS NITROGEN SUPPLY TEMPORARY MODIFICATION, PER EC-43821, RCS / ECCS FLOW REQUIREMENTS FOR THE INTACT LOOP, UNDER NATURAL CIRCULATION AND/OR ECCS FLOW DELIVERY WOULD INITIALLY SHEAR ANY INFILTRATING N2 INTO THE FLOW STREAM FOR EITHER THE FAULTED OR INTACT LOOP. WESTINGHOUSE LETTER LTR-LIS-08-543 IDENTIFIED VOID SIZES IN EXCESS OF THOSE EVALUATED WOULD NOT RESULT IN ADVERSE CONSEQUENCES. WESTINGHOUSE LETTER CWTR3-13-26 CONCLUDED THAT THERE WOULD BE NO ADVERSE AFFECTS. FOR THE (2) HOUR PERIOD DURING WHICH ADDITIONAL N2 FLOW COULD CONTINUE INTO SIT 2B, INITIALLY THE N2 WOULD NOT INFILTRATE INTO THE RCS AND THE TOTAL ADDITIONAL N2 GAS WOULD NOT EXCEED 18,000 SCF. SINCE AN EXPECTED GAS TO WATER INTERFACE AND FRACTION OF THE GAS EXPECTED TO INFILTRATE THE RCS FLUID WOULD FURTHER RESTRICT THIS VOLUME, AND SINCE THIS VOLUME WOULD BE A SMALL FRACTION OF THE EXPECTED SIT INVENTORY (BOTH GAS AND GAS FRACTION FROM THE SATURATED WATER), WHICH WOULD IN TURN BE A SMALL FRACTION OF THE TOTAL WATER INVENTORY, THE EFFECT UPON THE RECIRCULATING WATER VOLUME OF THE ADDITIONAL N2 GAS WOULD BE INSIGNIFICANT AND WOULD NOT ADVERSELY AFFECT THE ECCS (INCLUDING THE CONTAINMENT SPRAY).

DURING THE LONG TERM RECOVERY FROM SMALL BREAK LOCA (SBLOCA) AND SOME MODERATE BREAK LOCA (MBLOCA) EVENTS, THE RECOVERY ACTIONS MAY INCLUDE USING THE SDC SYSTEM FOR LONG TERM COOLING, IN LIEU OF RAS. CURRENTLY OPERATING PROCEDURES AND THE DESIGN AND LICENSING BASIS FOR SDC OPERATION INCLUDES MAINTAINING A REDUCED SIT PRESSURE TO AVOID OVERPRESSURIZING THE SDC AND LOSING THE SDC FUNCTION. THIS MAY BE ACCOMPLISHED BY VENTING THE SIT, WHICH COULD ENTAIL REPEATED VENTING OF SIT 2B IF THE N2 SUPPLY REPRESSURIZES IT, AND/OR ISOLATING THE SITS FROM THE RCS BY THE CLOSING OF THE SIT ISOLATIONS (WHICH CAN REQUIRE RESTORING POWER TO THE SIT ISOLATION VALVES). COMMITMENT A3995 IDENTIFIED THAT A METHOD WOULD BE DEVELOPED TO PLACE SDC IN SERVICE WITH NO NEED FOR THE OPERATOR TO LEAVE THE CONTROL ROOM. THIS METHOD WOULD INCLUDE RESTORING POWER TO THE SIT ISOLATIONS, TO ISOLATE THE SITS FROM THE RCS, AND PREVENT THEIR INJECTION DURING RCS DEPRESSURIZATION. THIS COMMITMENT (A3995) IS UNAFFECTED BY THIS TEMPORARY MODIFICATION. ISOLATING THE N2 HEADER IN (2) HOURS FOLLOWING THE RESPECTIVE DBA WOULD ALLOW NORMAL TREATMENT OF THE SIT 2B FOR SDC OPERATION, IN THAT ADDITIONAL OPERATOR SCRUTINY OF SIT 2B PRESSURE AND ADDITIONAL REPEATED VENTING OF SIT 2B (BEYOND THAT WHICH WOULD OCCUR NORMALLY TO SUPPORT SDC OPERATIONS), DURING SDC OPERATION AS PART OF THE RECOVERY ACTIONS, TO COPE WITH THE CONTINUOUS N2 INJECTION, PER THE TEMPORARY MODIFICATION (EC-43821), WOULD NOT BE REQUIRED.

THE ADDITION OF THE TEMPORARY MODIFICATION AND ASSOCIATED PIPING/TUBING, ESP. THE NNS PORTION OF THE CHANGE WILL BE EVALUATED UNDER EC-43821 FOR SEISMIC AND PIPE STRESS. THE NON NUCLEAR SAFETY RELATED NITROGEN SUPPLY PIPING INSIDE CONTAINMENT IS NOT CONSIDERED HIGH ENERGY PIPING PER UFSAR SECTION 3.6 AND 3.6A THEREFORE THE TEMPORARY MODIFICATION WILL NOT BE EVALUATED FOR PIPE WHIP OR SUBEQUENT JET IMPINGEMENT. CONSIDERATION OF OTHER SYSTEMS IMPACTING THE TEMPORARY MODIFICATION, VIA JET IMPINGEMENT (E.G. MAIN STEAM), AND UNDESIRED CONSEQUENCES IN THE EVENT OF A BREAK OF THE NITROGEN SUPPLY INSTALLED PER THIS MODIFICATION IS EVALUATED WITHIN THE EC-43821, AND THIS WILL ENSURE THAT UNDESIRED CONSEQUENCES TO SSC IMPORTANT TO SAFETY, AS A RESULT OF JET IMPINGEMENT UPON THE TEMPORARY MODIFICATION WILL NOT RESULT.

CURRENTLY SAFETY INJECTION SYSTEM OPERATING PROCEDURE OP-009-008 IDENTIFIES THAT (2) SIT TANK NITROGEN PRESSURE ADDITIONS CANNOT BE MADE SIMULTANEOUSLY (COMMITMENT P24648). THIS PASSIVE COMMITMENT WAS ESTABLISHED IN RESPONSE TO NRC INFORMATION NOTICE IN 96-031, WHICH IDENTIFIED THAT IT MAY BE POSSIBLE TO DEPRESSURIZE (2) SIT TANKS THROUGH THE SAME FAULT, IF THEY WERE CROSS-CONNECTED VIA THE N2 FILL SYSTEM, AT THE TIME OF A LIMTING ACCIDENT (LOCA). BASED ON THE ARRANGEMENT OF THE CONTINUOUS NITROGEN ADDITION

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PIPING IN EC-43821, WHICH INCLUDES (2) SAFETY RELATED CHECK VALVES, INSTALLED IN SERIES IN THE SIT 2B, THE CONTINUOUS N2 ADDITION PIPING PER EC 43821 PROVIDES SIT 2B A CONTINUOUS NITROGEN SUPPLY *IN LIEU OF* USING THE NORMAL NITROGEN FILL VALVE NG-162B. THEREFORE SIT 2B IS NOT VULNERABLE TO THE SAME FAILURE AS IDENTIFIED IN IN-96-031, SINCE, WHEN CROSSCONNECTING THE NITROGEN FILL PIPING. WHEN N2 ADDITIONS TO OTHER SIT TANKS ARE MADE IN CONCERT WITH/DURING THE CONTINUOUS OPERATION OF TEMPORARY MODIFICATION EC-43821, THE DUAL CHECK VALVES WILL PRECLUDE A FAILURE OF NON-SAFETY RELATED PIPING ASSOCIATED WITH SIT2B. ALSO THE FILL PIPING INSIDE THE DUAL CHECK VALVES IS SAFETY RELATED AND SEISMICALLY QUALIFIED. THEREFORE IT WILL NOT BE NECESSARY TO ISOLATE THE CONTINUOUS N2 FILL PIPING TO SIT 2B (AS SHOWN ON EC-43821), OR TO ENTER TS 3/4.5.1, TO PERFORM A NITROGEN ADDITION TO SIT 1A, 1B, OR 2A, WHILE THE SIT 2B CONTINUOUS FILL IS IN SERVICE.

NOTE THAT THE PIPING/TUBING/HOSES ADDED PER THE TEMPORARY MODIFICATION ARE DESIGNED, SEISMICALLY QUALIFIED, AND INSTALLED SUCH THAT THESE COMPONENTS WILL NOT FAIL, AND ADD TO THE SAFETY INJECTION SUMP DEBRIS LOADING (SEE EC-43821).

THEREFORE THERE WILL NOT BE AN INCREASE IN THE LIKELIHOOD OF OCCURRENCE OF A MALFUNCTION OF AN SSC, IMPORTANT TO SAFETY, PREVIOUSLY EVALUATED IN THE UFSAR. NOTE THAT THE RESPONSE TO QUESTION 3 ALSO INCLUDES INFORMATION PERTINENT TO THE RESPONSE TO QUESTION 2.

Question 3. Will the change Result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR? NO

BASIS:

PER 10CFR50 APPENDIX K, "ECCS EVALUATION MODELS"; "THE EFFECTS ON REFLOODING RATE OF THE COMPRESSED GAS IN THE ACCUMULATOR WHICH IS DISCHARGED FOLLOWING ACCUMULATOR WATER DISCHARGE SHALL ALSO BE TAKEN INTO ACCOUNT". PER WESTINGHOUSE LETTERS LTR-LIS-08-543 AND CWTR3-13-26 THE NITROGEN DISCHARGED INTO THE RCS, POST LOCA, WAS EVALUATED:

WESTINGHOUSE LETTER LTR-LIS-08-543 QUALITATIVELY EVALUATED THE IMPACT OF THE EXISTING SIT NITROGEN COVER GAS DISCHARGE (NOT INCLUDING THE CONTINUOUS GAS ADDITION OF EC-43821) UPON ECCS DELIVERY DURING ACCIDENT CONDITIONS. THE EVALUATION CONSIDERS THE EFFECT UPON HEAT TRANSFER WITHIN THE RCS, THE EFFECT UPON ECCS DELIVERY TO MITIGATE CORE UNCOVERY, AND THE IMPACT OF COLLECTION OF GAS VOIDS WITHIN THE RCS AND IMPACT UPON ECCS PERFORMANCE. THE EVALUATION CONSIDERS THESE EFFECTS UPON LARGE BREAK LOCA (LBLOCA), SMALL BREAK LOCA (SBLOCA), LONG TERM CORE COOLING (LTCC), AND LOCA FORCES (WATERHAMMER). LTR-LIS-08-543 IS ATTACHED FOR REFERENCE (NOTE THAT THIS DOCUMENT IS WESTINGHOUSE PROPRIETARY). THIS 50.59 WILL NOT REITERATE THIS EVALUATION, HOWEVER THE LETTER CONCLUDED THAT: 1.) DURING SBLOCA FOR WESTINGHOUSE AND CE DESIGNED NSSS WITH RECIRCULATING STEAM GENERATORS); "THE INJECTION OF THE NON-CONDENSIBLE GAS VOID VOLUME OF INTEREST DURING COLD LEG INJECTION IS NOT EXPECTED TO IMPACT THE CORE COOLING RESPONSE DURING A SBLOCA". THE GAS VOLUMES OF INTEREST ARE THOSE LISTED IN TABLE 1 OF THE LETTER. 2.) DURING LBLOCA (FOR ALL DESIGNS); "THE ADDITION OF THE EXTRA GAS VOLUME TO THE RCS WILL HAVE NO IMPACT IN THE SHORT TERM IN THIS REGARD. GAS THAT IS INJECTED PRIOR TO ACCUMULATOR EMPTY TIME IS EJECTED THROUGH THE BREAK VIA ECCS BYPASS OR WILL HAVE NEGLIGIBLE EFFECTS UPON CORE HEAT TRANSFER SINCE A LARGE AMOUNT OF VAPOR MAY ALREADY BE PRESENT IN THE CORE REGION AT THAT TIME. IN THE LONGER TERM, THE MAJORITY OF THESE GASES EITHER MIGRATE TO THE UPPER SPACES OF THE RCS OR OUT INTO CONTAINMENT WHERE THEY HAVE NO BEARING ON CORE COOLING RESPONSE BECAUSE THE CORE IS IN A STRATIFIED, BOILING MODE OF HEAT TRANSFER, WITH THE MIXTURE RESIDING IN THE AREA OF THE HOT/COLD LEG PENETRATIONS", AND 3.) "IT SHOULD BE NOTED THAT ALL OF THE ABOVE ARGUMENTS FOR LOCAS ARE CONSIDERED APPLICABLE TO GAS VOLUMES LARGER THAN THOSE IDENTIFIED IN TABLE 1".

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WESTINGHOUSE LETTER CWTR3-13-26 WAS PROVIDED TO SPECIFICALLY EVALUATE THE POTENTIAL FOR THE CONTINUOUS NITROGEN SUPPLY INSTALLED PER TEMPORARY MODIFICATION EC-43821 TO OPERATE IN THE EVENT THAT VALVE NG-157 FAILS TO CLOSE, POST-ACCIDENT, AND THE EFFECT OF THE CONTINUOUS NITROGEN SUPPLY TO SIT 2B UPON THE RCS AND ECCS ACCIDENT RECOVERY MODEL(S). CWTR3-13-26 IS ATTACHED TO THIS EVALUATION, AND FORMS A PART OF THE EVALUATION IN RESPONDING TO THE QUESTIONS. NOTE THAT CWTR3-13-26 IS WESTINGHOUSE PROPRIETARY. CWTR3-13-26 ASSUMES A CONTINUOUS 150 SCFM FEED TO SIT 2B, DUE TO THE ASSUMED FAILURE OF NG-157 TO CLOSE, WHICH CORRESPONDS WITH A 0.18 LBM/SEC. THE LETTER IDENTIFIES THAT DURING THE CORE REFLOOD PERIOD A MINOR, AND IN-SIGNIFICANT, INCREASE IN CORE HYDRAULIC RESISTANCE WOULD RESULT FROM THE CONTINUOUS NITROGEN SUPPLY BETWEEN ~51 SECONDS AND ~103 SECONDS (SEE FIGURE 2 OF THE LETTER). HOWEVER, CWTR3-13-26 CONCLUDED THAT THE 0.18 LBM/SEC IS INSIGNIFICANT COMPARED WITH THE 42.6 LBM/SEC AVERAGE NITROGEN INJECTION (NOT INCLUDING THE CONTINUOUS NITROGEN FLOW). THE INCREASE IN THE HYDRAULIC RESISTANCE K-FACTOR IS NEGLIGBLE, THUS THE ANTICPIATED INCREASE IN THE LOOP RESISTANCE TO REFLOOD, FROM 103 SECONDS AFTER CONTACT UNTIL THE END OF THE TRANSIENT IS NEGLIGIBLE ("IT IS JUDGED THAT THE EQUIVALENT K-FACTOR WILL NOT INCREASE BY MORE THAN ~0.1 FROM THE CURRENT VALUE OF ~260"). HENCE THE IMPACT ON REFLOOD RATE, PEAK CLAD TEMPERATURE (PCT), PEAK LOCAL OXIDATION (PLO), AND CORE WIDE OXIDATION (CWO) ARE JUDGED TO BE NEGLIGIBLE. CWTR3-13-26 (ATTACHMENT 3) ALSO STATES: "THE INCREASE IN NITROGEN MASS IN THE SITS DUE TO THE CONTINUOUS NITROGEN INJECTION DURING THE TIME OF THE LBLOCA IS ALSO NEGLIGIBALE. THE INITIAL NITROGEN MASS IN THE SITS IS 2243 LBM. THE INCREASE IN NITROGEN MASS DUE TO THE CONTINUOUS NITROGEN INJECTION DURING THE 1ST 100 SECONDS TO THE TRANSIENT (TIME WHEN THE SITS EMPTY) IS 18 LBM OF NITROGEN. THIS INCREASE HAS NO IMPACT ON THE RESULTS". SEE THE ATTACHED LETTER CWTR3-13-26 (ATTACHMENT 3) FOR FURTHER INFORMATION. BASED UPON THE CONCLUSIONS OF WESTINGHOUSE LETTER CWTR3-13-26, THE IMPACT UPON THE LOCA ANALYSES IS EITHER NEGLIGIBLE OR THERE IS NO IMPACT DUE TO THE TEMPORARY MODIFICATION (& CONTINUOUS N2 FEED). THERE IS NO ADVERSE IMPACT UPON LONG TERM COOLING (LTCC), DUE TO EITHER THE REMOTE MANUAL ISOLATION OF THE NITROGEN SUPPLY ISOLATION, VALVE NG-157, OR IN THE EVENT THAT NG-157 FAILS TO CLOSE, BY MANUALLY ISOLATING NITROGEN GAS HEADER ISOLATION NG-151 OR NG-152 (PER ATTACHMENT 1 TO THIS EVALUATION). NOTE THAT DURING SBLOCA (I.E. BREAK SIZES < 0.05 SQUARE FEET), FOR WHICH NATURAL CIRCULATION OCCURS, THE SITS WILL NOT FULLY DISCHARGE INTO THE RCS, AND THE ADDITION OF CONTINUOUS NITROGEN PER THE TEMPORARY MODIFICATION WILL HAVE NO EFFECT. FOR LARGER BREAK SIZES, WHEN THE SITS WOULD FULLY DISCHARGE, AND DURING WHICH NATURAL CIRCULATION WOULD NOT BE CREDITED, THE WESTINGHOUSE LETTER, CWTR3-13-26 (ATTACHMENT 3) EVALUATES THE EFFECT OF THE ADDITIONAL GAS AND DETERMINES THERE TO BE NO EFFECT OR THAT THE EFFECT IS NEGLIGIBLE. (WESTINGHOUSE LETTER FOR BREAK SIZE LIMITS APPLICABLE TO THE LETTER CWTR3-13-26 R/1, ATTACHMENT 3)

NOTE THAT FOR THE PURPOSES OF THIS EVALUATION, AND IAW 10CFR50 APPENDIX K, SECTION I.D.1; SINGLE FAILURE CRITERION, AN ANALYSIS OF POSSIBLE FAILURE MODES OF ECCS EQUIPMENT & OF THEIR EFFECTS ON ECCS PERFORMANCE MUST BE MADE .... ASSUMING THE MOST "DAMAGING SINGLE FAILURE OF ECCS EQUIPMENT HAS TAKEN PLACE". FOR THE PURPOSES OF THIS EVALUATION THAT SINGLE FAILURE IS ASSUMED TO BE THE SINGLE ACTIVE FAILURE OF NG-157 TO CLOSE AUTOMATICALLY UPON CIAS. DUE TO THE INCONSEQUENTIAL BUT NEGATIVE EFFECT UPON RCS HYDRAULIC RESISTANCE DURING THE REFLOOD SEQUENCE (PER CWTR3-13-26 R/1), THE CONTINUOUS NITROGEN SUPPLY AND THE MASS ADDITION TO THE RCS IS EVALUATED FOR ITS IMPACT TO THE ECCS DELIVERY UPON REFLOODING THE CORE.

NOTE THAT THE TEMPORARY MODIFICATION IS EVALUATED FOR ITS CONSEQUENCES UPON RCS, CORE THERMAL-HYDRAULIC PERFORMANCE AND ECCS DELIVERY, AND THE RESULT HAS BEEN DETERMINED TO BE NEGLIGIBLE. THEREFORE THERE WILL BE NO INCREASE IN THE RADIOLOGICAL CONSEQUENCES OF ANY ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR AS A RESULT OF IMPLEMENTING THIS TEMPORARY MODIFICATION.

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THE IMPACT UPON THE CONTAINMENT BLDG PEAK PRESSURE, OF THE N2 ADDITON BEING RELEASED INTO THE CONTAINMENT ATMOSPHERE, POST-LOCA, IS ALSO INCONSEQUENTIAL: THE CONTAINMENT PEAK PRESSURE OCCURS VERY EARLY IN THE ACCIDENT SEQUENCE, AT LESS THAN 20 SECONDS, WHICH IS INSUFFICIENT TO ALLOW ANY SIGNIFICANT NITROGEN ADDITION VIA THE TEMPORARY MODIFICATION (150 SCFM/60 * 15 = 37.5 SCF. USING A PEAK PRESSURE OF 37.1 PSIG, WOULD RESULT IN 14.9 CUBIC FEET OF NITROGEN AT PEAK CONTAINMENT PRESSURE, IN A TOTAL CONTAINMENT VOLUME OF 2.575 X 10^6 CUBIC FEET). THE EFFECT ON CONTAINMENT PEAK PRESSURE REDUCTION WITHIN 24 HOURS IS ALSO INSIGNIFICANT: AT AN ASSUMED CONSTANT PEAK TEMPERATURE OF 260 DEGREES F (THE ACCIDENT ANALYSIS PEAK TEMPERATURE IS < 260 DEGREES F);

MIN CONTAINMENT VOLUME: 2.575 X 10^6 CUBIC FEET

TOTAL N2 MASS INJECTED 0.18 LBM/SEC X 7200 SECONDS = 1296 LBM.

(1296LBM / 2,575,000) X (55.15 FT-LBF/LBM-^{O}R) X 720 ^{O}R X 1/144 = 0.14 PSI PRESSURE ADDITION AFTER 24 HR

PEAK PRESSURE AT 24 HOURS IS 16 PSIG (PER CALCULATION ECS98-015, CONTAINMENT P&T RESPONSE ANALYSIS - STEAM GENERATOR REPLACEMENT PROJECT) + 0.14 = 16.14 PSIG < 1/2 OF 37.1 PSIG. THE RESULTS SHOW THAT THE 24 HOUR PRESSURE IS LESS THAN ONE HALF OF THE CONTAINMENT PEAK PRESSURE.

THEREFORE THE CONTINUOUS NITROGEN SUPPLY WILL HAVE NO IMPACT UPON PEAK CONTAINMENT PRESSURE. CONTAINMENT PENETRATION CB MPEN014 IS PROTECTED INSIDE CONTAINEMNT WITH SAFETY CLASS 2 CONTAINMENT ISOLATION (CHECK) VALVE NG-158, AND OUTSIDE CONTAINMENT WITH SC-2 CONTAINMENT ISOLATION VALVE NG-157. NG-157 IS A FAIL CLOSED ISOLATION VALVE THAT CLOSES ON CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS) (REFERENCE UFSAR TABLE 6.2-32). FOR PURPOSES OF ASSUMING A LIMITING SINGLE FAILURE FOR THE CHANGE BEING EVALUATED, NG-157 IS ASSUMED TO FAIL TO CLOSE, TO CONTINUE SUPPLYING NITROGEN. TO PROTECT AGAINST A CONTAINMENT EGRESS, CHECK VALVE NG-158 WOULD CLOSE ON HGH DOWNSTREAM PRESSURE, IN THE EVENT OF A FAILURE OF THE NG PIPING INSIDE CONTAIMMENT, AND A FAILURE OF NG-157 TO CLOSE.

THEREFORE THE CHANGE PER EC-43821 WILL NOT RESULT IN MORE THAN A MINIMAL INCREASE IN THE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR.

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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: AS DISCUSSED IN QUESTION 3 ABOVE, SI IPT0343 WILL BE REMOVED FROM SERVICE. HOWEVER, SI IPT0343 IS NOT CREDITED IN THE UFSAR FOR POST-ACCIDENT MONITORING. REMOVING SI IPT0343 (SIT 2B NARROW RANGE PRESSURE) FROM SERVICE RESULTS IN A LOSS OF ONE OF THREE SIT 2B PRESSURE INDICATORS. SI IPT 0343 PROVIDES CONTROL ROOM INDICATION OF SIT 2B NARROW RANGE (350-750 PSIG) PRESSURE ALONG WITH CORRESPONDING ANNUNCIATION OF LO-LO (610 PSIG) AND HI-HI(660 PSIG) TANK PRESSURE AND COMPUTER POINT A44405. HOWEVER, LOSS OF THIS INDICATION DOES NOT INCREASE THE CONSEQUENCES OF A LOCA. SI IPT0342 (SIT 2B NARROW RANGE PRESSURE) WILL STILL BE AVAILABLE TO PROVIDE NARROW RANGE PRESSURE INDICATION AND LO (615 PSIG) AND HI (655 PSIG) TANK PRESSURE ANNUNCIATION TO ASSIST OPERATIONS IN MAINTAINING TANK PRESSURE WITHIN TECHNICAL SPECIFICATION LIMITS. ADDITIONALLY THE FUNCTION OF SI IPT0341 (SIT 2B WIDE RANGE PRESSURE) WHICH IS CREDITED IN THE FSAR FOR ACCIDENT MONITORING WILL NOT BE AFFECTED. THEREFORE THE ABILITY OF OPERATIONS TO MONITOR SIT 2B PRESSURE FOR VERIFICATION OF TECHNICAL SPECIFICATION REQUIREMENTS AND ACCIDENT MONITORING IS ONLY MINIMALLY AFFECTED AND WILL NOT INCREASE THE CONSEQUENCES OF ANY ACCIDENT EVALUATED IN THE UFSAR.

IN THE EVENT THAT A LBLOCA OCCURS, AND A SINGLE FAILURE OF VALVE NG-157 TO CLOSE OCCURS, THE CONTINUOUS N2 FEED TO SIT 2B WILL CONTINUE FOR (2) HOURS UNITL THE NITROGEN HEADER IS MANUALLY ISOLATED (SEE ATACHMENT 1). THIS RESULTS IN THE POTENTIAL ADDITION OF 150 SCFM FOR TWO HOURS, OR 18,000 SCF TO SIT 2B, AND SUBSEQUENTLY, TO THE RCS UNDER EVALUATED DESIGN BASIS ACCIDENT (DBA) CONDITIONS. AS DISCUSSED IN QUESTION 3, ABOVE, THE CONSEQUENCES OF A FAILURE FO NG-157 TO CLOSE WOULD BE A NONCONSEQUENTIAL RISE IN THE RCS HYDRAULIC RESISTANCE (K-FACTOR), WHICH WOULD HAVE A NEGLIGIBLE IMPACT OR NO IMPACT UPON THE ACCIDENT ANALYSES. SEE QUESTION 3, ABOVE, AND ATTACHMENTS 2 AND 3, WESTINGHOUSE LETTERS LTR-LIS-08-543 AND CWTR3-13-26, ATTACHED TO THIS EVALUATION. ADDING AN OPERATOR ACTION TO ISOLATE THE NITROGEN SUPPLY DOES NOT ALTER ANY ASSUMPTIONS USED, OR AUTOMATIC ACTIONS CREDITED WITHIN THE ACCIDENT ANALYES, AND IS SPECIFIED TO ENSURE THAT LOONG TERM COOLING IS ACCOMPLISHED WITHOUT ANY ADVERSE FFECTS FROM THE CONTINUOUS NITROGEN ADDITION, AND TO VALIDATE THAT THERE WILL BE NO ADVERSE EFFECT UPON LONG TERM CONTAINMENT PEAK PRESSURE REDUCTION. NO GREATER RELIANCE UPON THE CONTAINMENT ISOLATION VALVES WILL BE REQUIRED DUE TO THE INSTALLATION OF THIS TEMPORARY MODIFICATION. THE CONTAINMENT ISOLATION VALVES, NG-157 AND NG-158 WILL PROVIDE PROTECTION FROM A CONTAINMENT EGRESS IN THE EVENT OF A LOSS OR BREAK OF THE NON SAFETY RELATED NITROGEN SUPPLY SYSTEM PIPING .

MAINTAINING THE TECHNICAL SPECIFICATION MAXIMUM SIT PRESSURE AT OR LESS THAN 670 PSIG (TS 3/4.5.1) WILL ENSURE THAT SAFETY ANALYSIS ASSUMED NITROGEN GAS ADDITION TO THE RCS IS MINIMIZED AND WITHIN SAFETY ANALYSIS ANALYZED/ASSUMED LIMITS AND ALSO TO PROVIDE MARGIN TO THE SIT TANK RELIEF VALVE (SI-328B) SETPOINT OF 700 PSIG. IF THE SIT 2B GAS PRESSURE CONTINUED TO BUILD IN (OVER AN EXTENDED TIME PERIOD), IT COULD BE POSSIBLE TO REACH THE RELIEF VALVE (SI-328B) SETPOINT, RESULTING IN THE RELIEF VALVE LIFTING, AND BLOWDOWN OF THE SIT. THIS IS NOT CONSIDERED A CREDIBLE EVENT SINCE THE CONTINUOUS N2 SUPPLY WILL ONLY BE SUPPLYING LESS THAN 1.5 CUBIC FOOT OF GAS PER MINUTE, AT THE NORMAL/MAXIMUM SUPPLY DIFFERENTIAL PRESSURE (AT THE SUPPLY RATE OF < 50 SCFM), AND THERE WILL BE APPROXIMATELY 750 CUBIC FEET OF GAS SPACE IN SIT 2B. IN ORDER TO RAISE THE SIT 2B PRESSURE > 670 PSIG WOULD REQUIRE > 8 HOURS OF UNMONITORED GAS ADDITION, AND TO REACH THE RELIEF VALVE SETPOINT WOULD EXCEED 24 HOURS OF UNMONITORED GAS ADDITION. SURVEILLANCES CONDUCTED PER TS SURVEILLANCE PROCEDURE OP-903-001 WILL PRECLUDE THIS FROM OCCURRING. THEREFORE THERE IS NO INCREASE IN THE CONSEQUENCES OF A MALFUNCTION OF AN SSC, IMPORTANT TO SAFETY, PREVIOUSLY EVALUATED IN THE UFSAR, AFFECTED BY THIS CHANGE.

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NOTE THAT THE CONNECTION OF WIDE RANGE PRESSURE TRANSMITTER SI IPT341, AND NARROW RANGE TRANSMITTER, SI IPT342, IS MADE TO WHAT IS NOW THE N2 SUPPLY CONNECTION AT SIT2B. THIS CONNECTION IS ADJACENT TO THE SIT AND THEREFORE WOULD NOT BE ADVERSELY AFFECTED BY THE N2 FLOW. TESTING IN EC-43821 WILL INCLUDE MEASUREMENT OF PRESSURE FLUCTUATIONS CAUSED BY THE NITROGEN FEED. THEREFORE NO IMPACT TO UNCERTAINTY OR PRESSURE READINGS AT THE TRANSMITTERS DUE TO FLOW EFFECTS OF THE NITROGEN SUPPLY WILL RESULT.

THE IMPACT UPON THE CONTAINMENT BLDG PEAK PRESSURE, OF THE N2 ADDITON BEING RELEASED INTO THE CONTAINMENT ATMOSPHERE, POST-LOCA, IS ALSO INCONSEQUENTIAL: THE CONTAINMENT PEAK PRESSURE OCCURS VERY EARLY IN THE ACCIDENT SEQUENCE, AT LESS THAN 20 SECONDS, WHICH IS INSUFFICIENT TO ALLOW ANY SIGNIFICANT NITROGEN ADDITION VIA THE TEMPORARY MODIFICATION (150 SCFM/60 * 15 = 37.5 SCF. USING A PEAK PRESSURE OF 37.1 PSIG, WOULD RESULT IN 14.9 CUBIC FEET OF NITROGEN AT PEAK CONTAINMENT PRESSURE, IN A TOTAL CONTAINMENT VOLUME OF 2.575 X 10^6 CUBIC FEET). THE EFFECT ON CONTAINMENT PEAK PRESSURE REDUCTION WITHIN 24 HOURS IS ALSO INSIGNIFICANT: AT AN ASSUMED CONSTANT PEAK TEMPERATURE OF 260 DEGREES F (THE ACCIDENT ANALYSIS PEAK TEMPERATURE IS < 260 DEGREES F);

MIN CONTAINMENT VOLUME: 2.575 X 10^6 CUBIC FEET

TOTAL N2 MASS INJECTED 0.18 LBM/SEC X 7200 SECONDS = 1296 LBM.

(1296LBM / 2,575,000) X (55.15 FT-LBF/LBM-^{O}R) X 720 ^{O}R X 1/144 = 0.14 PSI PRESSURE ADDITION AFTER 24 HR

PEAK PRESSURE AT 24 HOURS IS 16 PSIG (PER CALCULATION ECS98-015, CONTAINMENT P&T RESPONSE ANALYSIS - STEAM GENERATOR REPLACEMENT PROJECT) + 0.14 = 16.14 PSIG < 1/2 OF 37.1 PSIG. THE RESULTS SHOW THAT THE 24 HOUR PRESSURE IS LESS THAN ONE HALF OF THE CONTAINMENT PEAK PRESSURE.

THEREFORE THE TEMPORARY MODIFICATION RESULTS IN LESS 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: THE TEMPORARY MODIFICATION PER EC-43821 WILL USE CONTAINMENT ISOLATION VALVE NG MVAAA157 TO PROVIDE A CONTINUOUS FEED OF NITROGEN (N2) TO SAFETY INJECTION TANK (SIT) 2B. CURRENTLY NG-157 IS LISTED IN THE UFSAR TABLE 6.2-32 AS NORMALLY OPEN & CLOSES UPON RECEIPT OF A CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS). THE TEMPORARY MODIFICATION PER EC-43821 DOES NOT CHANGE THE IDENTIFIED NORMAL, ACCIDENT OR SHUTDOWN IDENTIFIED POSITIONS FOR NG-157 PER UFSAR TABLE 6.3-32. SINCE THE TEMPORARY MODIFICATION WILL FACILITATE A CONTINUOUS FEED OF N2 THROUGH NG-157 TO SIT 2B, FOR UP TO (2) HOURS POST DBA, CONSIDERATION OF THIS CHANGE UPON THE STRUCTURE SYSTEMS AND COMPONENTS (SSC) AFFECTED BY THE NITROGEN SYSTEM AND SIT2B, WILL INCLUDE EVALUATION WHETHER AN ACCIDENT OF A DIFFERENT TYPE THAN THAT CONSIDERED IN THE UFSAR MAY BE INTRODUCED.

THE BULK NITROGEN STORAGE SYSTEM IS NOT NORMALLY ALIGNED TO SIT 2B FOR CONTINUOUS NITROGEN SUPPLY. THE BULK NITROGEN SYSTEM IS DESIGNED TO PROVIDE N2 MAKEUP TO THE SITS, AS NEEDED, DURING NORMAL PLANT OPERATION. THIS TEMPORARY MODIFICATION CHANGES THE N2 SUPPLY TO SIT 2B TO A CONTINUOUS SUPPLY. SINCE THE NITROGEN HEADER CONTAINMENT ISOLATION VALVE, NG-157, IS NORMALLY OPEN, AND CLOSES UPON RECEIPT OF A CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS), THE POSSIBILITY OF A SINGLE ACTIVE FAUILURE OF NG-157 IS CONSIDERED AS A LIMITINING SINGLE FAILURE FOR PUSPOSES OF THIS EVALUATION. NG-157 IS IDENTIFIED ON FLOW DIAGRAM G166 SHEET 1 AS SAFETY CLASS 2, FAILS CLOSED, AND RECEIVES ASSOCIATED SAFETY RELATED TRAIN "B" POWER SUPPLY. THE ASSOCIATED LINE NUMBER FOR THE NG-157 CONTAINMENT PENETRATION, UP TO THE INSIDE CONTAINMENT ISOLATION NG-158 (SC-2, CHECK VALVE PER G166 SHT. 1) IS 2NG1-47. THIS PIPING IS IDENTIFIED WITH A DESIGN PRESSURE OF 800 PSIG AND AN OPERATING PRESSURE OF 700 PSIG (PER INDUS PASSPORT, NON VALIDATED FIELDS). SC-2 RELIEF VALVE NG-1523 IS PROVIDED WITH AN 800 PSIG SETPOINT AT A SUFFICIENT CAPACITY TO PROTECT AGAINST A FAILURE OF THE UPSTREAM N2 REGULATING STATION AT NG-147A(B) AND NNS RELIEF VALVE NG-149 (REF. ER-W3-98-0665 & PER INDUS VALIDATED FIELD) TO ENSURE THAT THE BULK NITROGEN STORAGE SYSTEM CANNOT

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OVERPRESSURIZE THE ASSOCIATED SYSTEM COMPONENTS, INCLUDING CONTAINMENT PENETRATION CB MPEN014 (AT NG-157 AND NG-158. THE BULK N2 SUPPLY SYSTEM PROVIDES A HIGHLY RELIABLE ESSENTIAL BACKUP GAS SOURCE FOR USE IN CHARGING VARIOUS SAFETY RELATED ACCUMULATORS. THE CONTINUOUS N2 SUPPLY TO SIT 2B PER EC-43821 WILL BE CONNECTED TO THE N2 SYSTEM DOWNSTREAM OF NORMALLY CLOSED, NON-NUCLEAR SAFETY RELATED, VALVE NG-502. THE TEMPORARY MODIFICATION WILL UTILIZE A HOSE CONNECTION AND A (NON-NUCLEAR SAFETY RELATED) REGULATOR TO FURTHER REDUCE THE N2 SUPPLY PRESSURE TO A NOMINAL 635 PSIG, TO MAINTAIN THE SIT 2B N2 PRESSURE WITHIN THE DESIRED RANGE. TWO (2) SAFETY RELATED CHECK VLAVES WILL BE INSTALLED IN SERIES IN THE SAFETY RELATED PORTION OF THE N2 SUPPLY DOWNSTREAM OF THE NNS REGULATOR. THE PORTION OF THE TEMPORARY MODIFICATION UP TO THE DUAL CHECK VALVES WILL BE SAFETY RELATED, SINCE IT WILL BE IN CONTINUOUS OPERATION AND COMMUNICATING WITH THE SIT 2B COVER GAS VOLUME, WHICH IS DESIGNATED SAFETY CLASS 2 (PER G167 SHT. 2 AND UFSAR 3.9). THE (2) CHECK VALVES IN SERIES WILL PRECLUDE A LOSS OF SIT 2B PRESSURE IN THE EVENT OF A FAILURE OF THE UPSTREAM NON-SAFETY-RELATED PIPING, AND IN THE EVENT OF A SINGLE ACTIVE FAILURE OF ONE OF THE CHECK VALVES TO CLOSE. PER THE UFSAR SECTION 3.6 AND 3.6A, THE NITROGEN GAS PIPING INSIDE THE CONTAINMENT, INCLUDING THE NON-SAFETY RELATED PORTION, IS NOT CONSIDERED HIGH ENERGY PIPING AND IS NOT EVALUATED FOR PIPE WHIP OR SUBSEQUENT JET IMPINGEMENT, THEREFORE THESE CONDITIONS ARE NOT CONSIDERED CREDIBLE FOR THE TEMPORARY MODIFICATION. NOTE THAT THE FLEXIBLE HOSE IN THE NON-SAFETY RELATED PORTION OF THE SIT 2B NITROGEN FILL TEMPORARY MODIFICATION WILL BE SECURED TO ADJACENT GRATING TO AFIX THE HOSE AND PROVIDE SOME MEASURE OF PROTECTION AGAINST UNDESIRED MOVEMENT. THE SAFTEY RELATED PIPING AND CONNECTING NON-SAFETY RELATED PIPING IS EVALUATED FOR SEISMIC AND PIPE STRESS EFFECTS (SEE EC 43821).

SIT 2B IS PROVIDED WITH A SAFETY CLASS 2 SAFETY RELIEF VALVE, SI MVAAA328B (UFSAR TABLE 3.9-10) WHICH IS SET AT 700 PSIG AND HAS A DISCHARGE CAPACITY (GAS) OF 1120 SCFM (UFSAR 6.3.2.2.4.1) PER CALCULATION EC-M98-006, SIT 2B SAFETY RELIEF VALVE SI-328B HAS A DISCHARGE CAPACITY OF 6429.98 SCFM. THE MAXIMUM N2 FLOW RATE THAT CAN BE SUPPLIED TO SIT 2B, IS 150 SCFM NITROGEN FLOW RATE (PER EC-43821). THEREFORE IN THE EVENT OF A FAILURE OF THE N2 BULK SYSTEM SUPPLY REGULATORS (NG-147A AND B) & THE N2 BULK SYSTEM SAFETY RELIEF VALVE (SAFETY RELIEF VALVE, NG-149), IN CONCERT WITH A FAILURE (PRESSURE REGULATING FAILURE) OF THE REGULATOR INSTALLED PER EC-43821, THE MAXIMUM N2 FLOW WILL NOT EXCEED THE CAPACITY OF THE SIT 2B SAFETY RELIEF VALVE, SI-328B. THEREFORE SIT 2B WILL NOT EXCEED ITS 700 PSIG DESIGN PRESSURE (REFERENCES: DESIGN BASIS DOCUMENT W3-DBD-001, UFSAR SAFETY RELIEF VALVES TABLE 3.9-10, UFSAR SECTION 3.9, CALCULATION EC-M98-006).

THE SAFETY INJECTION TANKS' WATER VOLUMES ARE DESIGNED TO HAVE A MAINTAINED COVER OF NITROGEN GAS PRESSURE, IN ORDER TO PRESSURIZE AND PROVIDE A MOTIVATING FORCE TO THE SIT WATER VOLUME, TO DISCHARGE THAT WATER VOLUME TO THE REACTOR COOLANT SYSTEM, DURING AND FOLLOWING A LOSS OF COOLANT ACCIDENT (LOCA). THE TEMPORARY MODIFICATION PER EC-43821 PROVIDES A CONTINUOUS FEED OF THE SAME COVER GAS (N2) FROM THE BULK NITROGEN STORAGE UNIT, TO ENSURE THE GAS COVER PRESSURE IS MAINTAINED WHILE MINIMIZING THE OPERATOR BURDEN AND THE ASSOCIATED PLANT RISK RESULTING FROM EXCESSIVE N2 MAKEUP EVOLUTIONS TO SIT 2B. THE ADDITION OF CONTINUOUS NITROGEN COVER GAS TO SIT 2B PER EC-43821 DOES NOT ALTER THE DESIGN OF THE SAFETY INJECTION TANK. THE ADDITION OF THE CONTINUOUS N2 FEED, ASSUMING A FAILURE OF NG-157 TO CLOSE, UPON RECEIPT OF A CIAS, WILL NEITHER OVERPRESURIZE THE SIT 2B NOR ALLOW A FAILURE OF THE CONNECTING NON-SAFETY RELATED PIPING TO RESULT IN A FAILURE NOT PREVIOUSLY EVALUATED. THE MAXIMUM POSSIBLE N2 FLOW RATE AND THE REDUNDANT CHECK VALVES INCLUDED IN THE DESIGN OF THE TEMPORARY MODIFICATION WILL PROVIDE SUFFICIENT SAFETY FEATURES, AS DESCRIBED, TO PRECLUDE THE POSSIBILITY OF AN ACCIDENT OF A DIFFERENT TYPE THAN PREVIOUSLY EVALUATED IN THE UFSAR.

SINCE THE AIR INSIDE CONTAINMENT IS COMPOSED OF > 75% NITROGEN, AND NITROGEN IS AN INERT GAS, THE ADDITION OF NITROGEN TO THE CONTAINMENT ATMOSPHERE IN THE EVENT OF A FAILURE OF THE NEW NNS NITROGEN PIPING/TUBING ADDED PER EC-43821 WOULD NOT RESULT IN ANY NEW FAILURE OR ACCIDENT NOT PREVIOUSLY EVALUATED IN THE UFSAR. DURING NORMAL PLANT OPERATION THE CONTAINMENT ATMOSPHERE MAY BE PERIODICALLY PURGED, WHICH WOULD EXHAUST ANY ACCUMULATION OF NITROGEN, AND THEREBY ENSURING A SATISFACTORY CONTAINMENT ATMOSPHERE WHEN A CONTAINMENT PERSONNEL ENTRY MAY BE

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REQUIRED, TO PRECLUDE A PERSONNEL SAFETY ISSUE.

THEREFORE THIS CHANGE, THE TEMPORARY MODIFICATION PER ERC-43821 DOES NOT CREATE THE POSSIBILITY OF 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 TEMPORARY MODIFICATION WILL SHUNT THE SENSING LINE FOR NARROW RANGE PRESSURE TRANSMITTER SI IPT0343, RENDERING THE TRANSMITTER INOPERABLE. THIS TRANSMITTER IS DESCRIBED IN THE UFSAR (SEE QUESTION 3, ABOVE), HOWEVER THIS TRANSMITTER IS NOT CREDITED FOR POST ACCIDENT MONITORING AND REDUNDANT NARROW RANGE PRESSURE TRANSMITTER SI IPT0342 WILL REMAIN IN SERVICE TO ENSURE THAT SIT 2B PRESSURE WILL REMAIN WITHIN TECHNICAL SPECIFICATION LIMITS. OP-903-001, TECHNICAL SPECIFICATION SURVEILLANCE LOGS, ATTACHMENT 11.20 IDENTIFIES USING NARROW RANGE TRANSMITTER SI IPT343 (PMC POINT ID A44405) FOR THE REQUIRED PRESSURE RANGE SURVEILLANCE VERIFICATION OF TS SURVEILLANCE 4.5.1.a, EVERY 12 HOURS, THAT THE SIT NITROGEN COVER PRESSURE IS WITHIN 600-670 PSIG (TS 3.5.1.d). THE REMAINING NARROW RANGE PRESSURE TRANSMITTER, SI IPT0342, WILL ALOW PERFORMANCE OF THE TECH SPEC SURVEILLANCE LOGS, AND SHOULD SI IPT0342 BECOME INOPERABLE, THE ASSOCIATED TECHNICAL SPECIFICATION SHUTDOWN ACTION (TS 3.5.1.b, 72 HOUR LCO) WILL BE ENTERED. THE REMAINING NARROW RANGE TRANSMITTER, SI IPT0342, WILL BE CONNECTED VIA A JUMPER TO THE WIDE RANGE TRANSMITTER, SI-IPT-0341, SO THAT THE TUBING ASSIGNED TO ALLOW THE CONTINUOUS N2 SUPPLY, PER EC-43821, MAY BE ISOLATED FROM THE TRANSMITTER, THEREBY PREVENTING SPURIOUS PRESSURE READINGS DUE TO THE N2 FLOW EFFECTS (SEE DRAWINGS IN ERC -43821).

IN THE EVENT THAT CONTAINMENT ISOLATION VALVE NG-157 EXPERIENCES A SINGLE ACTIVE FAILURE TO CLOSE UPON RECEIPT OF A CIAS, AND/OR UPON REMOTE MANUAL ISOLATION, THE NITROGEN SUPPLY COULD CONTINUE UNINTERRUPTED UNTIL THE SUPPLY HEADER IS MANUALLY ISOLATED. SINCE THE N2 SUPPLY HAS AN 800 PSIG DESIGN PRESSURE - PROTECTED BY RELIEF VALVE NG-1523 (SEE ABOVE), AND THE SUPPLY HEADER/PIPING PRESSURE DROP WOULD LIKELY RESULT IN A LOWER N2 PRESSURE IN THE EVENT OF CONTINUOUS N2 FLOW, AND ASSUMING A FAILURE OF THE ADDITIONAL (NON-SAFETY RELATED) PRESSURE REGULATOR, ADDED PER EC-43821, IT IS POSSIBLE THAT A N2 SUPPLY PRESSURE OF UP TO 800 PSIG COULD BE SUPPLIED TO SIT 2B, AT A FLOW RATE OF UP TO 150 SCFM, FOLLOWING A FAILURE OF NG-157 TO CLOSE UPON RECEIPT OF A CIAS. FOLLOWING A DBA THAT WOULD RESULT IN A CIAS, OPERATOR ACTION TO MANUALLY ISOLATE THE NITROGEN GAS HEADER AT NG-151 OR NG-152 IN (2) HOURS PRECLUDES SUFFICIENT NITROGEN INFILTRATION INTO THE RCS TO RESULT IN A MALFUNCTION OF ASSOCIATED COMPONENTS (SEE ABOVE BASIS FOR RESPONSE TO QUESTION 2). IN THE EVENT OF A FAILURE OF RELIEF VALVE NG-1523, IN CONJUNCTION WITH A FAILURE OF THE NNS NITROGEN REGULATING STATION (NG-147A(B) AND NNS RELIEF VALVE, NG-149), THE CONTINOUS NITROGEN SUPPLY COULD PRESSURIZE SIT 2B UP TO THE RELIEF VALVE SETPOINT OF 700 PSIG (SI-328B). THE SIT 2B RELIEF VALVE, SI-328B, HAS SUFFICIENT DISCHARGE CAPACITY, OF UP TO 6429.98 SCFM, TO PREVENT SIT 2B OVERPRESSURIZATION IN THIS EVENT (CALC ECM98-006). VALVE NG-157 IS IMPORTANT TO SAFETY SINCE IT PROVIDES A CONTAINMENT ISOLATION FUNCTION IN THE EVENT OF A DESIGN BASIS ACCIDENT AND RECEIPT OF A CIAS. AS SUCH, NG-157 IS CLASSIFIED AS SAFETY CLASS 2 (DWG. G-166 SHT. 1), IT FAILS IN THE CLOSED POSITION, AND ITS CLOSED SAFETY FUNCTION IS PROVIDED WITH A REDUNDANT BACKUP BY INSIDE CONTAINMENT ISOLATION VALVE, NG-158 (A SAFETY CLASS 2 CHECK VALVE), TO PROTECT AGAINST A CONTAINMENT EGRESS (G166 SHT. 1, UFSAR SECTION 3.9). IN THE EVENT NG-157 FAILED TO CLOSE DURING OR FOLLOWING A DESIGN BASIS ACCIDENT, UPON RECEIPT OF A CIAS, THIS FAILURE COULD RESULT IN AN UNINTERRUPTED NITROGEN FEED TO SIT 2B, SINCE NG-158 ONLY PROTECTS AGAINST CONTAINMENT EGRESS CONDITIONS, AND NO REDUNDANT PROTECTION AGAINST A CONTINUED INGRESS OF NITROGEN EXISTS IN THE EVENT OF A SINGLE ACTIVE FAILURE OF NG-157 TO CLOSE. ALTHOUGH THE RESULT OF A MALFUNCTION OF NG-157, IN FAILING TO CLOSE COULD RESULT IN A CONTNUOUS FEED OF NITROGEN TO SIT 2B, DURING AND FOLLOWING A DESIGN BASIS ACCIDENT, THIS CONDITION WOULD NOT CAUSE A DIFFERENT RESULT THAN ANY PREVIOUSLY ANALYZED IN THE UFSAR (INCLUDING SAFETY ANALYSES).

DURING AND FOLLOWING A DESIGN BASIS ACCIDENT, SUCH AS A LARGE BREAK LOSS OF COOLANT ACCIDENT

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(LBLOCA), THE SAFETY INJECTION TANKS ARE DESIGNED TO PROVIDE A PASSIVE SYSTEM THAT AUTOMATICALLY DISCHARGES INTO THE COLD LEGS OF THE REACTOR COOLANT SYSTEM (RCS), BORATED WATER, TO PROVIDE CORE COOLING AND REFILL (OR REFLOOD), WHEN THE RCS PRESSURE FALLS BELOW THE SIT PRESSURE. THE N2 COVER GAS PROVIDES AN ELASTIC MOTIVATING FORCE, THAT WILL FORCE THE BORATED WATER INTO THE RCS, THEREBY MAKING UP TO THE REACTOR, SUFFICIENT WATER VOLUME (IN CONJUNCTION WITH HIGH PRESSURE SAFETY INJECTION, AND AT LOWER RCS PRESSURE, LOW PRESSURE SAFETY INJECTION) TO RE-COVER THE REACTOR CORE, THEREBY MINIMIZING THE PEAK FUEL CLAD TEMPERATURE EXCURSION, AND PROVIDING ADDITIONAL FISSION POISON AND DECAY HEAT REMOVAL. THE REFLOOD SEQUENCE OF THE EMERGENCY CORE COOLING SYSTEM (ECCS) DELIVERY IS SENSITIVE TO THE RCS PRESSURE IN THAT THE RCS PRESSURE MUST BE LESS THAN THE MINIMUM SIT TANK PRESSURE BEFORE THE SIT WILL BEGIN FLUID DELIVERY. AT THE ONSET OF FLUID DELIVERY THE PRESENCE OF ADDITIONAL NITROGEN SUPPLY TO SIT 2B, IN THE EVENT A SINGLE ACTIVE FAILURE OF NG-157 TO CLOSE OCCURS, WOULD BE BENEFICIAL, SINCE THE NITROGEN WOULD ASSIST IN THE ECCS DELIVERY. DURING A LBLOCA, THE SIT TANKS ARE ASSUMED TO DELIVER THEIR ENTIRE INVENTORY TO THE RCS, AND THE RCS BREAK ASSUMED WOULD RESULT IN A COMPLETE DEPRESSURIZATION AND BLOWDOWN OF THE RCS TO THE CONTAINMENT SUCH THAT THE SITS WOULD INJECT THEIR ENTIRE FLUID VOLUMES, AS WELL AS A QUANITITY OF THE NITROGEN DRIVING GAS, WHICH WILL BE ON THE ORDER OF THOUSANDS OF CUBIC FEET, ONCE EXPANDED TO THE CONTAINMENT BACK PRESSURE (REFERENCE WESTINGHOUSE LETTER LTR-LIS-08-543). ASSUMING THE FULL DEPRESSURIZATION OF THE RCS AND DELIVERY OF THE SIT 2B NITROGEN COVER GAS TO THE RCS, THE PIPING SIZE PROVIDED IN THE TEMPORARY MODIFICATION (EC-43821), ALONG WITH EXISTING N2 PIPING AND TUBING, WILL LIMIT THE MAXIMUM N2 FLOW RATE TO < 150 SCFM, ASSUMING A DELIVERY PRESSURE OF 800 PSIG AND A DEPRESSURIZED SIT 2B (SEE EC-43821 FOR MAXIMUM CALCULATED CONTINUOUS FLOW TO SIT 2B).

THE CONTINUOUS NITROGEN SUPPLY, FOLLOWING A MALFUNCTIONOF NG-157 (SINGLE FAILURE TO CLOSE) IS EVALUATED FOR THE EFFECTS UPON THE RCS AND THE EXISTING ACCIDENT ANALYSES PER WESTINGHOUSE LETTER CWTR3-13-26, WHICH DETERMINED THAT THE NITROGEN ADDITION WOULD EITHER NOT HAVE A SIGNIFICANT AFFECT UPON, OR NOT ADVERSELY AFFECT, EXISTING ANALYSES. WESTINGHOUSE LETTER CWTR3-13-26 WHICH EVALUATED THE ADDITION OF NITROGEN VIA THE CONTINUOUS NITROGEN SUPPLY, FOLLOWING A FAILURE OF NG-157 TO CLOSE, IDENTIFIED THAT A "NEGLIGIBLE" INCREASE IN RCS LOOP HYDRAULIC RESISTANCE TO ECCS DELIVERY FROM 103 SECONDS TO THE END OF THE TRANSIENT ANALYSIS WOULD NOT IMPACT FUEL PEAK CLAD TEMPERATURE. THESE LETTERS ARE PROVIDED AS ATTACHMENTS 2 AND 3 TO THIS EVALUATION (NOTE THAT THESE LETTERS ARE WESTINGHOUSE PROPRIETARY).

THE ADDITION OF NITROGEN GAS FROM THE TEMPORARY MODIFICATION, IN THE EVENT THAT CONTAINMENT ISOLATION VALVE NG-157 EXPERIENCED A SINGLE ACTIVE FAILURE TO CLOSE, UPON OTHER SYSTEMS (OTHER THAN THE RCS) IS EVALUATED, TO DETERMINE IF THERE IS ANY IMPACT TO EXISTING SAFETY ANALYSES, DIFFRENT FROM ANY PREVIOUSLY EVALUATED: THE CONTAINMENT PEAK PRESSURE, BOTH FOR IMPACT TO PEAK PRESSURE AND FOR REDUCTION IN PEAK PRESSURE TO LESS THAN ONE HALF AFTER 24 HOURS IS PROVIDED IN RESPONSE TO QUESTION 4. FOR IMPACT TO THE ECCS, POST ACCIDENT, SUFFICIENT NON-GAS SATURATED WATER WILL BE PRESENT AND MIXING WITHIN THE RCS, PRIOR TO DISCHARGE TO THE SAFETY INJECTION SUMP, TO PRECLUDE A MALFUNCTION OF THE ECCS, AND A LOSS OF THE ECCS OR CONTAINMENT SPRAY, AS EVALUATED IN THE RESPONSE TO QUESTION 2.

A MALFUNCTION OF ONE OF THE TWO SAFETY RELATED CHECK VALVES ADDED BY EC-43821 WOULD NOT ADVERSELY AFFECT THE SIT 2B, SINCE THE REDUNDANT CHECK VALVE WOULD CLOSE AND PREVENT SIT 2B FROM DEPRESSURIZING.

A MALFUNCTION OF THE WIDE RANGE INSTRUMENT, SI IPT341, AND THE CONSEQUENCES OF THE MALFUNCTION IS NOT AFFECTED BY THE TEMPORARY MODIFICATION PER EC-43821. THIS INSTRUMENT IS USED FOR POST ACCIDENT SIT PRESSURE MONTORING, TO ENSURE SIT 2B HAS DISCHARGED ITS INVENTORY AND THE PRESSURE MEAUSUREMENT, INCLUDING REFERENCE LEG ACCURACY WILL NOT BE ADVERSELY AFFECTED BY THE TEMPORARY MODIFICATION. IN THE EVENT OF A LOSS OF PRESSURE INDICATION VIA MALFUNCTION OF NARROW RANGE INSTRUMENT SI-IPT342, THIS WOULD PREVENT MEETING THE SURVEILLANCE REQUIREMENT PER TS 4.5.1.A, EVERY 12 HOURS, AND THE ASSOCIATED SHUTDOWN ACTION STATEMENT WOULD BE ENTERED, SINCE SIT 2B PRESSURE (NARROW RANGE) COULD NOT BE ADEQUATELY VERIFIED. SIT 2B PRESSURE COULD BE DETERMINED VIA SI-IPT341 FOR THE SHORT TERM, HOWEVER THE INSTRUMENT FUNCTION (OR MALFUNCTION) DOES NOT HAVE

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ANY DIRECT INTERACTION WITH THE BEHAVIOR OF SIT 2B OR THE SAFETY ANALYSES.

NOTE THAT THE POST SBLOCA ACCIDENT RECOVERY WOULD INCLUDE ALIGNMENT AND OPERATION OF THE SHUTDOWN COOLING SYSTEM (SDC) FOR REACTOR DECAY HEAT REMOVAL. TO PREVENT OVERPRESSURIZATION OF THE SDC SYSTEM, THE SIT PRESSURE MUST BE REDUCED TO < 300 PSIG. PER UFSAR 6.3.2.2.1 THE SIT PRESSURES WILL BE REDUCED "BY THE OPERATOR" TO BETWEEN 235 AND 300 PSIG, TO PREVENT INADVERTANT PRESSURIZATION OF THE SDC DURING PLANT COOLDOWN, WHEN THE RCS PRESSURE IS LESS THAN 1750 PSIA. THIS MAY BE ACCOMPISHED BY CLOSING NG-157. IN THE EVENT THAT NG-157 FAILS TO CLOSE THE OPERATOR CAN PERFORM A MANUAL ISOLATION OF NG-151 OR NG-152 (SEE ATTACHMENT 1). ISOLATING NG-151 OR NG-152 WILL NOT INTERRUPT NITROGEN TO ANY SSC REQUIRED TO PERFORM A POST ACCIDENT SAFETY FUNCTION. THE BULK NITROGEN GAS SUPPLY IS NOT DESIGNED AS A CONTINUOUS FEED, AND USING THE SYSTEM TO PROVIDE A CONTINUOUS N2 SUPPLY TO SIT 2B IS CONTINGENT UPON THE ABILITY TO MANUALLY ISOLATE THE N2 HEADER WITHIN 2 HOURS FOLLOWING A DESIGN BASIS ACCIDENT, IN THE EVENT THAT NG-157 FAILS TO ISOLATE, AND/OR PRIOR TO SDC OPERATION TO PREVENT AN INADVERTANT SDC OVERPRESSURIZATION.

THE ADDITION OF THE TEMPORARY MODIFICATION AND ASSOCIATED PIPING/TUBING, ESP. THE NNS PORTION OF THE CHANGE WILL BE EVALUATED UNDER EC-43821 FOR SEISMIC AND PIPE STRESS. THE NON NUCLEAR SAFETY RELATED NITROGEN SUPPLY PIPING INSIDE CONTAINMENT IS NOT CONSIDERED HIGH ENERGY PIPING PER UFSAR SECTION 3.6 AND 3.6A THEREFORE THE TEMPORARY MODIFICATION IS NOT REQUIRED TO BE EVALUATED FOR PIPE WHIP OR SUBSEQUENT JET IMPINGEMENT.

THEREFORE A MALFUNCTION THE TEMPORARY MODIFICATION PER EC-43821 WILL NOT INTRODUCE A DIFFERENT RESULT IN THE SAFETY ANALYSES/UFSAR THAN PREVIOUSLY EVALUATED.

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 EFFECTS OF VOIDING FROM THE SAFETY INJECTION TANKS DISCHARGING THEIR COVER GAS INTO THE RCS IS EVALUATED IN WESTINGHOUSE LETTER LTR-LIS-08-543 (ATTACHED), WHICH CONCLUDES THAT THERE IS NO ADVERSE EFFECT UPON THE RCS AND REACTOR POST ACCIDENT THERMNAL HYDRAULIC PERFORMANCE. SEE ATTACHMENT 2. THE CONTINUOUS NITROGEN SUPPLY IS EVALUATED PER WESTINGHOUSE LETTER CWTR3-13-26 FOR IMPACT TO RCS AND REACTOR CORE THERMAL-HYDRAULIC PERFORMANCE. ATTACHMENT 3 DETERMINES THAT THE EFFECT TO CORE REFLOOD, PEAK FUEL CLAD TEMPERATURE (PCT), PEAK LOCAL OXIDATION (PLO), & CORE WIDE OXIDATION (CWO) IS NEGLIGIBLE, AND CONCLUDES THAT NO ADVERSE EFFECT TO THESE PARAMETERS WILL RESULT. SEE ATTACHMENT 3.

DURING NORMAL PLANT OPERATION, THE TEMPORARY MODIFICATION PROVIDES A CONTNUOUS COVER GAS SUPPLY TO SIT 2B THROUGH CONTAINMENT ISOLATION VALVES NG-157 AND NG-158. NG-157 IS IDENTIFIED AS NORMALLY OPEN, FAIL CLOSED, SAFETY CLASS 2 (PER G166-SHT 1, UFSAR 3.9, AND OP-003-019). NG-158 IS A CHECK VALVE PROVIDING INSIDE CONTAINMENT ISOLATION FROM CONTAINMENT EGRESS (UFSAR 3.9). THE CONTAINMENT ISOLATION VALVES ARE NOT ALTERED BY THIS TEMPORARY MODIFICATION, AND IN THE EVENT OF A NNS PIPING FAILURE THESE VALVES WILL PRECLUDE A CONTAINMENT EGRESS THROUGH THE ASSOCIATED CONTAINMENT PENETRATION. THE AFFECT UPON THE CORE THERMAL PERFORMANCE POST-LOCA AND THEREFORE UPON THE EFFECT UPON FUEL AND THE PEAK CLAD TEMPERATURE OF GAS VOIDING AS A RESULT OF THE DISCHARGE OF THE SIT TANKS' COVER GAS INTO THE RCS DURING AND FOLLOWING A DESIGN BASIS ACCIDENT IS DISCUSSED IN THE ATTACHED LETTER, LTR-LIS-08-543. THE CONTINUOUS NITROGEN SUPPLY IS EVALUATED PER WESTINGHOUSE LETTER CWTR3-13-26 FOR IMPACT TO RCS AND REACTOR CORE THERMAL-HYDRAULIC PERFORMANCE TO ENSURE PEAK FUEL CLAD TEMPERATURE (PCT), PEAK LOCAL OXIDATION (PLO), & CORE WIDE OXIDATION (CWO) IMPACT IS NEGLIGIBLE, AND CONCLUDES THAT NO ADVERSE EFFECTS WILL RESULT.

THE BASIS FOR RESPONSE TO QUESTION 4 IDENTIFIES THAT CONTAINMENT PEAK PRESSURE AND PEAK PRESSURE

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REDUCTION WITHIN 24 HOURS ARE NOT ADVERSELY AFFECTED BY THE TEMPORARY MODIFICATION.

THEREFORE THE TEMPORARY MODIFICATION PER EC-43821 WILL NOT RESULT IN A DESIGN BASIS LIMIT FOR A FISSION PRODUCT BARRIER BEING 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 TEMPORARY MODIFICATION DOES NOT IMPLEMENT NOR INTRODUCE ANY ANALYSIS METHODS AND DOES NOT ALTER ANY ANALYSIS METHODS CURRENTLY PROVIDED IN THE UFSAR. SAFETY ANALYSES FOR DBAS (CHAPTER 15 AND CHAPTER 6) AND OTHER ANALYSES PROVIDED IN THE UFSAR ARE UNAFFECTED BY THE TEMPORARY MODIFICATION PER EC-43821.

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.

Attachments:

Attachment 1: Review of Operator action to isolate the nitrogen gas header per ANS 58.8.
Attachment 2: Westinghouse Letter LTR-LIS-08-543
Attachment 3: Westinghouse Letter CWTR3-13-26 Revision 1

I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # / Rev. #: 2010-06 / R-1**

Proposed Change / Document: EC-14765, SI-405A(B) Bypass Fill / Equalization Line Addition / ECN-25944, Changes for Calculation MPR-2390 R3 SDC Gas Intrusion Analysis

Description of Change:

EC-14765 / ECN-25944 will install a fill / pressure equalization system for the Shutdown Cooling (SDC) system to compress air voids and minimize pressure transient events, which may occur when SI-405A(B), RC Loop SDC Suction Inside Containment Isolation, is opened. The new fill system will consist of a 3/4" diameter by-pass line installed around SI-405A(B) (RC Loop SDC Suction Inside Containment Isolation). The by-pass line will contain a flow control orifice and a normally closed solenoid valve SI-4052A(B) (RC Loop SDC Suction Inside Containment Bypass Isolation) that will be opened remotely from the Main Control Room (MCR) or remote shutdown panel (LCP-43) prior to opening SI-405A(B). The bypass line will allow Reactor Coolant to slowly fill and compress the SDC piping void and equalize pressure across valve SI-405A(B), preventing the previously experienced pressure transients when valve SI-405A(B) is opened. The proposed bypass line solenoid valve can be operated from either the Main Control Room or the Remote Shutdown Panel (LCP-43).

Shutdown Cooling Suction Isolation Valves, SI-405A(B) are required for containment isolation and are the class boundary separating the Shutdown Cooling Class 1 line from the Class 2 Low Pressure Safety Injection (LPSI) pump suction piping. These valves form part of the Interfacing System LOCA (ISL) boundary. EC-14765 / EC-25944 installs a bypass line around these valves with a new solenoid valve SI-4052A(B), which will perform the same containment isolation and pressure boundary design functions as the existing valves SI-405A(B) warranting this 50.59 Evaluation. The new solenoid valves will also duplicate the SI-405A(B) valves Open Permissive Interlock (OPI) that prevents valve opening with RCS pressure greater than ~~386~~ 392 psia (Shutdown Cooling entry conditions), and are fail closed on a loss of power.

Solenoid valves SI-4052A(B) require addition to Technical Specification Table 3.4-1 (Reactor Coolant System Pressure Isolation Valves) per LO-LAR-2010-0054 and will be listed as Containment ~~isolation~~ Isolation Valves in UFSAR Table 6.2-32.

The new piping and valves associated with the proposed fill / equalization line will be procured and installed as Safety Related in accordance with the ASME requirements for Class 1 and 2 piping and components. Electrical circuits have been designed to meet the Appendix R requirements. General Design Requirements (GDC), Environmental and Accident Conditions (UFSAR Table 3.11-1), and Seismic requirements have been considered and are met by the proposed fill / equalization system.

ECM03-003 (MPR2390 R-3) "Analysis of Waterford Station, Unit 3 SDC System Gas Accumulation", evaluates acceptable SDC system performance for the reactor coolant system (RCS) with a gas void downstream of Containment Isolation Valves SI-405A(B) at reactor containment penetrations 40 and 41 respectively. The calculation ECM03-003 considers the system pressure transient and resultant effect on LTOP pressure relief valves SI-406A(B) along with the NEI Guidelines for "Prevention and Management of System Gas Accumulation" (NEI 09-10 revision 1), for the SDC system and low pressure safety injection (LPSI) pump(s) A(B) performance.

ECM03-003 (MPR2390 R-3) determined: 1) that an orifice would be required to control the rate of downstream piping fill, 2) that it could take up to 9 minutes to completely collapse and pressurize the void in the downstream piping in order to control the pressure transient such that the possibility of lifting the Low Temperature Overpressure Protection (LTOP) relief valve was eliminated, 3) that during certain normal shutdown conditions,

[1] 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 high point gas void requires venting, 4) that during Design Basis Events (DBE), venting is not required and SDC flow degradation is tolerable and of very short duration, and 5) that void size acceptance criteria in OP-903-026, "Emergency Core Cooling System Valve Lineup Verification" should be revised.

The evaluation and analyses in revised calculation ECM03-003 concludes and incorporates the following:

- The gas void in the SDC suction piping will not affect LPSI system operation during injection mode since the gas void will remain isolated with the LPSI system in service.
- For normal shutdowns (non design basis events), the gas void at the system high point will require venting at valve SI-4051A(B) prior to opening valve SI-407A(B) and a LPSI pump start when reactor coolant system (RCS) pressure is less than 85 psia (with instrument uncertainty - 100 psia PMC / 110 psia board indicator) to meet the NEI guidance.
- Design basis accident conditions will not allow venting. Therefore further analysis has been performed in ECM03-003 which concludes the LPSI pumps flow may degrade to 71% of the desired 4100 gpm flowrate for approximately two minutes as the gas passes through the pump. The flow rate will return to the desired 4100 gpm without damaging or air binding the pump.
- For a failure of bypass / equalization solenoid valves SI-4052A(B) to open, valves SI-405A(B) can be opened with RCS pressure below 230 psia (with instrument uncertainty: 213 psia PMC / 203 psia board indicator) without causing a pressure transient to lift relief valves SI-406A(B) or causing excessive dynamic loads on the piping.
- The RCS system leak rate through valves SI-401A(B), SI-405A(B) and SI-4052A(B) needs to be less than 0.26 gal/min (train A) and 0.28 gal/min (train B). The existing programs for void detection and reactor coolant leakage will ensure meeting this requirement. OP-903-026 "Emergency Core Cooling System Valve Lineup Verification", requires inspections below SI-407A(B) for void detection at 31 day intervals. If a void is discovered the required UT will ensure the void is within the evaluated size. OI-040-000 "Reactor Coolant System Leakage Monitoring" requires investigation when unidentified leakage exceeds 0.1 gpm. A statement will be added to attachment 6.6 "An unidentified leak rate value greater than 0.26 gpm in conjunction with continued voiding below valve SI-407A(B) may be indicative of leakage past SI-401A(B).
- OP-903-026 R-17 "Emergency Core Cooling System Valve Lineup Verification" will require revision of sections 7.6 and 7.7 to remove the introduction of a gas void requirement as the SI-4052A(B) bypass valve will prevent the hydraulic transient and relief valve lifting. The water levels reflected in sections 6.11 and 6.12 should be revised to reflect the acceptable void sizes for system operability determination. Per ECM03-003 (MPR-2390 R-3) Sect. 3.2.1 using the conservative RWSP 55 degree temperature, Train A water level (gas void) may extend 21.5 ft. below the disc of valve SI-407A, Train B water level (gas void) may extend 25.1 ft. below the disc of valve SI-407B.
- Isometric drawing ESSE-SI205 and ESSE-SI206 along with flow diagram G-167 sheet 2 are revised to reflect the addition of an orifice coupling upstream of valve SI-4052A(B) to assure the bypass piping flow rate and resultant downstream pressure transient remains below the set point of the LTOP relief valves SI-406A(B). This flow control / limit results in a 10 minute fill / equalization time limit.

Administrative controls to open the breakers supplying power to SI-405A(B) and SI-4052A(B) valves during normal operation is included in OP-009-005, "System Operating Procedure Shutdown Cooling". This provides protection from opening these containment isolation valves (with no automatic isolation function) during normal plant operation. Included is the closing of the breakers supplying power to SI-405A(B) and SI-4052A(B) valves when valve manipulation is required.

UFSAR Chapter 9 EC Markup is revised to delete obsolete information pertaining to the hydraulic operators previously evaluated and replaced under EC 935 for the SI-405A(B) valves.

W3-DBD-1 EC markup is revised to include the ECM03-003 conclusions and recommendations and remove the SI-405A(B) closure stroke time requirement (ref. CR-WF3-2010-3645) as no design basis requirement for the valve closure time limit exists.

Summary of Evaluation:

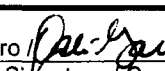
This evaluation determined that installation and operation of the SDC isolation pressure equalization bypass system to compensate for air voiding downstream of SI-405A(B) does not represent any unreviewed safety question and does not require prior NRC review and approval, except as already covered by License Amendment Request LO-LAR-2010-0054 (W3F1-2010-0019) discussed in Safety Evaluation 2010-06, Rev.0.

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.

The proposed change requires NRC approval due to the addition of the new solenoid valves SI-4052A(B) to Technical Specification Table 3.4-1 (Reactor Coolant System Pressure Isolation Valves). See license amendment request LO-LAR-2010-0054 (W3F1-2010-0019).

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

Preparer: Dale V. Gallodoro /  / Entergy / Design Engineering / 9-11-13
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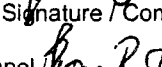
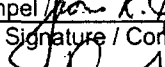
Reviewer: Thomas R. Hempel /  / Entergy / Design Engineering / 9-12-13
Name (print) / Signature / Company / Department / Date

OSRC: Brian Lanka /  / 9/18/13
Chairman's Name (print) / Signature / Date

W3 13-17
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:

The relevant accident for this modification is a loss of coolant accident (LOCA). During normal plant operations the new bypass line is isolated from the RCS by closure of the SI-401A(B) valve. The SI-401A(B) valve is tested for RCS leakage through the valve seat when closed. The SDC system is not generally considered an accident initiator due to the isolation of SDC until cold shutdown conditions are attained, at which time SDC can be placed into service. The SDC system bypass fill / pressurization line connects the piping the space around the SI-405A(B) valve. A new Class 1 solenoid valve, SI-4052A(B) isolates the piping in parallel with SI-405A(B). The SI-4052A(B) valves are interlocked with the same Open Permissive Interlock (OPI) as the SI-401A(B) and SI-405A(B) valves, such that the valves cannot be opened until pressurizer pressure is below 392 psia.

The addition of the fill / bypass line decreases the likelihood of a pressure transient occurring when SI-405A(B) is opened. The pressure transients were caused by the fast opening of SI-405A(B) because of the rapid collapse of an air void downstream of SI-405A(B). The transient has caused actuation of the Low Temperature Overpressure Protection (LTOP) relief valve with an isolatable loss of reactor coolant, which was quickly recognized by the Operations staff, who stopped the loss by closing SI-405B. (Ref. CR-WF3-2008-4161) During operation, the new solenoid valve in the bypass line will be closed and will have power removed by opening the breaker for the SI-4052A(B) control circuit. The control circuit has been designed to eliminate the likelihood of an intersystem LOCA through use of the same RCS Pressure interlock for both parallel valves. This modification adds about 10 feet of ¾" piping to each train of the shutdown cooling suction piping inside containment. Because the new piping and valves are designed, analyzed and qualified to the same ASME Section III Class 1 and 2 requirements as the existing piping and valves, this change does not result in more than a minimal increase in the frequency of occurrence of a LOCA. In addition, the valves are designed and qualified to meet containment isolation requirements.

The proposed modification credits administrative controls to hold the new bypass solenoid valve, SI-4052A(B), open for 10 minutes prior to opening SI-405A(B), to prevent the hydraulic transient. Calculation ECM03-003 shows that opening SI-4052A(B) for at least 9 minutes prior to opening SI-405A(B) collapses the air void such that a hydraulic transient capable of lifting the LTOP will not occur when SI-405A(B) is opened. Procedure OP-009-005 provides instructions to hold SI-4052A(B) open for a minimum of 10 minutes prior to opening SI-405A(B). The replacement combined control switch CS-2 on the remote shutdown panel (~~RSP~~LCP-43) is used to control both valves only during a main control room evacuation scenario. These valves cannot be operated from ~~this~~ either location as long as the RCS pressure interlock is functioning and power is removed at the DC distribution panel. The ~~interface pressure boundary~~pressure interlock feature also is part of the SI-401 valve function, and these valves are in series to provide ~~backup to each other~~a double isolation barrier. Since these valve interlocks remain functional and the use of these valves is not until cold shutdown, the combined control switch function does not result in a more than minimal frequency of occurrence of an accident previously evaluated in the

FSAR.

Accidents identified in the FSAR were evaluated with respect to the change proposed as described below. This change will have no significant ~~affect~~ effect on the entry to shutdown cooling as required in response to a Chapter 15 event or for normal plant shutdown. SI-4052A(B) have the same requirements as SI-405A(B).

FSAR Sections 15.1 and 15.2, Increase and decrease in heat removal by the secondary system (Turbine Plant)

This change installs a bypass line and valve in the shutdown cooling system. This change will have no impact on the main steam, feedwater or other secondary systems and will not impact the accidents analyzed for these systems.

FSAR Section 15.6, Decrease in reactor coolant system inventory

The impact of this change on the loss of coolant accident (LOCA) is relevant and is discussed above.

FSAR Section 15.7, Radioactive release from a subsystem or component & FSAR Section 15.9, Miscellaneous

This change installs a bypass line and valve in the shutdown cooling system, and will not impact the analyses presented in the FSAR.

The proposed modification will enhance the ability of the Shutdown Cooling system to perform its intended design function without the introduction of the previously experienced system pressure transients.

In summary, the design of the new bypass line and the administrative controls established ensure that the air void downstream of SI-405A(B) is collapsed or vented prior to placing SDC in standby and will prevent inadvertent opening of the LTOP relief valve. 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:

Because this change installs a new solenoid valve in parallel with the existing containment isolation valve SI-405A(B), the integrity of the containment boundary must be considered. This new valve, piping and supports are designed and qualified to ASME Section III requirements consistent with existing containment isolation valves. The valve meets the appropriate seismic and environmental qualification requirements. As new solenoid valves SI-4052A(B) are locked closed during plant operation, the requirement to stroke closed in a given time for containment isolation is not applicable. This is reflected in the EC markup of UFSAR Table 6.2-32 and is consistent with valves SI-405A(B). Consequently, this change does not result in more than a minimal increase in the likelihood of occurrence of an equipment malfunction.

The replacement combined control switch CS-2 on the remote shutdown panel (~~RSP~~LCP-43) is combined because of space limitations on the panel elevation such that the controls remain in the same functional location for these valves and their interface function with the related valves SI-401A(B) and SI-407A(B). To facilitate ~~the~~ two different valves operating from the same switch, opening the fill valve, SI-4052A(B), has been designed to require the operator to push in the switch knob instead of turning the switch knob as is normally done to open SI-405A(B). This allows the turning of the knob to be retained for the main valve SI-405A(B), the same as the

present switch is operated for the present valve. The push function to open is engraved on the indication position for open of the fill valve SI-4052A(B), which is the same location as the indication for this valve on the Main Control Room switch. Since the switch has only a push function, and no pull function is available, the standard use of pull to open cannot be used. Also, this limits the use of knob in-out to one function for the new valve. Closure of the fill valve is not necessary after the main valve is opened, except for isolation of the SDC line when both valves are required to be closed, so closure of the fill valve is part of closure of the main valve by the operator. The use of this switch is only during an evacuation of the MCR during a fire, where procedures will be closely followed, and is coupled with the fill valve indication clearly engraved with the push instruction. Additionally, the RCS pressure interlock remains in service when the valves are controlled from this ~~RSP~~LCP-43 location. Therefore, this replacement switch with dual valve control function is not more than minimally adverse to the frequency of a malfunction previously evaluated in the F SAR.

This SI-4052A(B) valve is normally locked closed along with the SI-405A(B) valve. Non-operation of this fail-closed valve will not affect the companion SI-405A(B) valve other than the control circuit which is in common, and therefore renders the train non-functional. This effect of loss of the shut down cooling train is no different than the present failure of the SI-405A(B) valve to operate, and Containment Isolation will be maintained because the valves fail closed and are locked closed already. Spurious operation is being addressed by converting the remote position indicating lights to be powered from the SUPS at 120 VAC. This allows the 125 VDC controls for the solenoids to be opened at the distribution panel breaker which de-energizes all conductors in the cables from having live DC power. The solenoids use DC power, and the only remaining energized conductors will have 120 VAC power for indication. DC powered solenoids will not operate with 120 VAC power applied, so spurious operation due to remote shorting will not occur within the cables. Therefore, there is no increase in occurrence of a malfunction not previously evaluated in the F SAR.

The proposed modification will enhance the ability of the Shutdown Cooling system to perform its intended design function eliminating the previously experienced system pressure transient. The pressure transient was caused by the rapid collapse of the known air void downstream of SI-405A(B) because of the fast opening of SI-405A(B). The transient has caused actuation of the Low Temperature Overpressure Protection (LTOP) relief valve, which was quickly recognized by the Operations staff, who stopped the loss of coolant by closing SI-405B.

The addition of the fill bypass line decreases the likelihood of a pressure transient occurring when SI-405A(B) is opened. The SDC suction gas accumulation is not caused by this modification. EC-14765 / ECN-25944 and Calculation ECM03-003 (MPR2390 R-3) only enhance the existing system and provide revised requirements to minimize the consequences of the already existing gas accumulation condition. The addition of the bypass line and the ECM03-003 (MPR2390 R-3) results with respect to the pressure transient, gas transport, RCS leakage requirement, and the gas void dynamic venting are addressed below.

The ¾" bypass fill line around valve SI-405A(B) is not aligned for LPSI SDC mode above 350F and 392 psig [UFSAR 6.3.2.9.7] plant mode 4. The loss of ¾" pipe integrity at the orifice with the LPSI in SDC service where the redundant train of SDC could be placed into service are bounded by existing analyses in the UFSAR for both large break and small break LOCAs. The frequency of a malfunction previously analyzed has not been increased by more than a minimal amount.

The performance of the LPSI / SDC system with a gas void between SI-405A(B) /SI-4052A(B) /SI-406A(B) and SI-407A(B) is evaluated in the ECM03-003 (MPR2390 R-3) calculation. The LPSI SDC tolerance to gas voiding and LPSI pump A(B) performance with administratively controlled gas volumes along with bounding LOCA operation gas volumes is presented in

ECM03-003 (MPR2390 R-3). The ECM03-003 (MPR2390 R-3) calculation provides system capability clarification with regards to operation and LTOP tolerance that does not adversely affect the likelihood of a malfunction of a structure, system, or component important to safety described in the UFSAR, and does not change the conclusions reached utilizing ECM03-003 revision 0. The ECM03-003 (MPR2390 R-3) calculation concludes that, should the SI-4052A(B) valve fail to open, SDC can be placed into service without lifting the LTOP relief valve by reducing RCS pressure to less than 230 psia and opening the SI-405A(B) valve. The saturation temperature for 230 psia is 394°F. To achieve the required 28°F subcooled margin for 230 psia and the temperature limit for SDC in OP-902-008, "Functional Recovery", T_H could be as high as 350°F, which is the normal SDC temperature limit. Therefore, reducing pressure to less than 230 psia would not require additional cooldown time to reach SDC entry conditions.

The ECM03-003 (MPR2390 R-3) gas transport analysis includes a bounding gas void passing through the SDC LPSI pumps for the limiting case that a gas void develops in the SDC reactor containment penetration piping coincident with a LOCA, or small break LOCA, requiring placement of the SDC into service without venting the gas void. The minor reduction of LPSI flow during the initial two minutes of placing the SDC system into service with the maximized gas void with an RCS pressure as low as 70 psia is analyzed to demonstrate the gas void does not gas bind the LPSI pump, though the maximized gas void would exceed the 5% gas void fraction NEI gas void guidance that is the current industry accepted limit. Administrative controls are in place to preclude the bounding gas void from forming. The existing administrative controls limit the size of potential gas voiding in the SDC reactor containment penetrations 40 and 41. The ECM03-003 (MPR2390 R-3) does not change conclusions reached from the revision 0 of the calculation. In addition, during the initial minutes of SDC operation, decay heat is still being removed by steaming through the Atmospheric Dump Valves. Therefore, there is no adverse impact on decay heat removal due to a short duration of degraded SDC flow.

The ECM03-003 (MPR2390 R-3) RCS leakage consideration requires administrative controls in plant instructions provide steps to verify gas void in the SDC reactor containment penetration 40 and 41 that supports the NEI 5% gas void fraction. The maximized gas void is analyzed to demonstrate the capability of the SDC system to perform beyond the allowable gas void conditions that we maintain in the SDC reactor containment penetration 40 and 41. During normal operations, when RCS pressure is below 392 psig, the bypass system will provide the means to pressurize potential gas voids in reactor containment penetration 40 and 41 by opening the SI-4052A(B) valve for ten minutes prior to opening SI-405A(B). The pressurization of potential gas void at penetration 40 and 41 will meet the 5% averaged gas entrainment guidance from NEI, while precluding lifting of the LTOP pressure relief valve SI-406A(B) when placing SDC into service. ECM03-003 (MPR2390 R-3) also recommends manually venting the SDC header inside containment after opening SI-401A(B) and SI-405A(B). This is a good practice for normal operation but is not required for design basis accidents where containment entry is not possible.

The ECM03-003 (MPR2390 R-3) gas void dynamic venting requires the SDC flow to exceed 2281 gpm to ensure the gas void is transported with the flow. Procedure OP-009-005, "System Operating Procedure Shutdown Cooling", is used to initiate shutdown cooling flow at 4100 gpm which exceeds the ECM03-003 (MPR2390 R-3) calculation requirements.

The existing two train design of SDC permits placement of the alternate SDC train into service should the SI-405A(B) valve fail to operate.

The likelihood of the bypass orifice becoming blocked is extremely remote due to the orifice 0.35" diameter and the RCS is maintained at a high degree of cleanliness though the monitoring of chemistry for detection of degradation material in the RCS combined with the charging system

feed-and-bleed through filtered pressurizer spray and letdown through the chemical volume control system.

The proposed modification credits administrative controls to hold the new bypass solenoid valve, SI-4052A(B), open for 10 minutes prior to opening SI-405A(B), to prevent the hydraulic transient. Calculation ECM03-003 shows that opening SI-4052A(B) for at least 9 minutes prior to opening SI-405A(B) collapses the air void such that a hydraulic transient capable of lifting the LTOP will not occur when SI-405A(B) is opened. Procedure OP-009-005 provides instructions to hold SI-4052A(B) open for a minimum of 10 minutes prior to opening SI-405A(B). The evaluation below addresses the ability to perform the new administrative control error free, concluding that the ability to recover from a foreseeable error and the time required are within the recovery time already evaluated for SI-401A(B). Therefore, the new administrative control does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The manual operator actions to close the power breakers to valves SI-405A(B) and SI-4052A(B) and to operate the SI-4052A(B) and SI-405A(B) valves in the proper sequence and timing were evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 when placing SDC system into service. The ten primary attribute evaluations were compared with the existing manual operator action to close the breakers to SDC valve SI-401A(B) and the same plant area location of the breakers for valves SI-405A(B) and SI-4052A(B):

(1) The specific operator actions required;

Specific operator action to close the breakers for valves SI-405A(B) and SI-4052A(B) is similar to those actions identified in OP-009-005 for closing breakers for the SI-401A(B) valves. The other new operator action is to open SI-4052A(B) for a minimum of 10 minutes prior to opening SI-405A(B). The controls for the new SI-4052A(B) valves are on the same panel as the SI-405A(B) valves in the control room and on the remote shutdown panel. Procedure OP-009-005, Shutdown Cooling, is updated to specify that SI-4052A(B) must be open for a minimum of 10 minutes prior to opening SI-405A(B). No adverse impact.

(2) The potentially harsh or inhospitable environmental conditions expected;

The environment for closing breaker for SI-405A(B) and SI-4052A(B) is the same as the existing action evaluated to close the breakers for the SI-401A(B) valves. The environment for operating SI-4052A(B) is the same as the existing operation of SI-405A(B). No adverse impact.

(3) A general discussion of the ingress/egress paths taken by the operators to accomplish functions;

The breakers for the SI-405A(B) and SI-4052A(B) valves are located in the same areas of the plant as the breakers for SI-401A(B). Ingress and egress paths for the SI-405A(B) and SI-4052A(B) valves are the same as taken to SI-401A(B). No adverse impact.

(4) The procedural guidance for required actions;

Existing procedural guidance for closure of SI-401A(B) valve breaker provides adequate template for operator guidance in the closure of the breaker for SI-405A(B) and SI-4052A(B) valves. Specific operator action to close the breakers for valves SI-405A(B) and SI-4052A(B) is similar to those actions identified in OP-009-005 for closing breakers for the SI-401A(B) valves. The other new operator action is to open SI-4052A(B) for a minimum of 10 minutes prior to opening SI-405A(B). The controls for the new SI-4052A(B) valves are on the same panel as the SI-405A(B) valves in the control room and on the remote shutdown panel. Procedure OP-009-005, Shutdown Cooling, is updated to specify that SI-4052A(B) must be open for a minimum of

10 minutes prior to opening SI-405A(B). No adverse impact.

(5) The specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions;

Operator training to perform specific action to close the breaker for SI-405A(B) and SI-4052A(B) valves is the same as that presently evaluated for SI-401A(B) breaker closure. Operator training will be provided for the specific actions specified in OP-009-005 for initiating SDC. No adverse impact.

(6) Any additional support personnel and/or equipment required by the operator to carry out actions;

The same operations personnel and equipment that address the closure of the SI-401A(B) valves and initiating SDC with the existing system can close the breaker switches for valves SI-405A(B) and SI-4052A(B) and initiate SDC using the modified system with the bypass line. The additional 10 minutes of time required to hold the SI-4052A(B) valve open prior to opening SI-405A(B) is acceptable because there are no specific design basis time requirements for initiating SDC, the actions are at least several hours into any DBE, and the additional 10 minutes of hold time will not significantly impact the Operator's ability to carry out other required actions. No adverse impact.

(7) A description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken;

The existing operation procedures that address placement of the Shutdown Cooling system into service identifies operator action and equipment needed for closing the SI-401A(B) breaker and for operating the SI-405A(B) valve. The action is expanded to include the SI-405A(B) and SI-4052A(B) valve breakers and operation of the SI-4052A(B) valve. No adverse impact.

(8) The ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery; and

The ability to recover from inadvertent action to close the breaker for SI-405A(B) and SI-4052A(B) valves is similar to the recovery presently evaluated for SI-401A(B) valve breaker closure. The recovery time is enveloped in the recovery time for closure of the SI-401A(B) valve breakers. The ability to recover from a credible failure to hold SI-4052A(B) open for at least 10 minutes prior to opening SI-405A(B) is assured based on operator training regarding actions to take in the event of an inadvertent LTOP relief valve actuation. Training demonstrates that operators can quickly recognized the loss of reactor coolant inventory through the SI-406B valve and respond to close SI-405A(B) before pressure level is depleted. The consequences would be bounded by the LOCA analysis. The recovery from a credible error in performance of the manual action evaluated here and the expected time to make such a recovery would not result in more than a minimal increase in consequences evaluated in the UF SAR. No adverse impact.

(9) Consideration of the risk significance of the proposed operator actions

The risk significance for action to close the breaker for SI-405A(B) and SI-4052A(B) valves is similar to the risk presently evaluated for SI-401A(B) valve breaker closure. The breakers for the SI-405A(B) and SI-4052A(B) valves are in the same plant areas as the breakers for the SI-401A(B) valves. No adverse impact.

(10) Time response as outline in ANSI/ANS-58.8-1994 [Reference 5]

The Shutdown Cooling system is not required for being placed into service at the beginning of a plant event. The later time sequence in the event for SDC system to be placed into service

affords the time for operator action to manually close the breaker for SI-405A(B) and SI-4052A(B) valves and to pause for 10 minutes between opening SI-4052A(B) and SI-405A(B). The time response is similar to the time evaluated for SI-401A(B) operator breaker closure action. No adverse impact.

With the application of the same safety classification of the SDC system to the bypass orifice and the installation of the same sized bypass orifice in each of the two redundant SDC trains there is not more than a minimal increase in the likelihood of a malfunction to a structure, system, or component important to safety previously evaluated in the UFSAR. Therefore, there is no increase in occurrence of a malfunction of an SSC 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:

The change proposed adds approximately 10 feet of ¾" diameter piping inside the containment building and is designed to meet reactor coolant pressure boundary requirements. The plant accidents were reviewed and found to be bounded by the LOCA analysis and Small Break LOCA analysis. These included plant accidents such as the main steam line break and steam generator tube rupture. The consequences of any failure of this piping, or the valves in the piping, are also bounded by existing analyses in the UFSAR for both large break and small break LOCAs. The new solenoid valve SI-4052A(B) will function in parallel with SI-405A(B) to provide a redundant high/low pressure interface and intersystem LOCA boundary with upstream valve SI-401A(B). The new piping and valves also meet the containment isolation / penetration requirements.

~~The new configuration will also not affect the entry to shutdown cooling conditions. SI-4052A(B) has the same requirements as SI-405A(B) and any failure of SI-4052A(B) would be bounded by that already analyzed for SI-405A(B).~~

The ECM03-003 (MPR2390 R-3) pressure transient analysis requires filling / pressurizing the air void downstream of SI-405A(B) for an additional 10 minutes prior to lining up the LTOP relief valve and placing the SDC system placement into service. The Waterford 3 Dose Analyses, for example, ECS04-013, "Small Break LOCA AST Radiological Dose Consequences", which is the bounding analysis for events where SDC could be initiated, credits reaching SDC entry conditions within 7.5 hours. A 10 minute delay in SDC service will not cause more than a minimal increase in the consequences of accidents evaluated in the UFSAR. Reaching SDC entry conditions with allowed cooldown rates is possible within only a few hours and so the additional 10 minutes of steaming while lining up SDC will not cause the 7.5 hours assumed in the analysis to be exceeded. With a SI-4052A(B) single failure, the other redundant train will remain available and capable of meeting the required safety function. ECM03-003 (MPR2390 R-3) evaluated that the failed SI-4052A(B) train may be placed in service below 230 psia with no pressure transient threat.

Thus, this modification does not increase the consequences of an accident previously evaluated in the UFSAR. Other accidents addressed in the FSAR are not impacted by this change as described in the response to Question 1 above.

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 change proposed adds a piping bypass system around existing Containment Isolation Valve (CIV) SI-405A(B) that becomes a part of the reactor coolant pressure boundary. The radiological release consequences associated with failure of the new ¾" diameter piping and valves are bounded by the existing failure analysis of the existing 14" shutdown cooling piping.

The performance of the LPSI / SDC system with a gas void between SI-405A(B) /SI-4052A(B) /SI-406A(B) and SI-407A(B) is evaluated in the ECM03-003 (MPR2390 R-3) calculation. The LPSI SDC tolerance to gas voiding and LPSI pump A(B) performance with administratively controlled gas volumes along with bounding LOCA operation gas volumes is presented in ECM03-003 (MPR2390 R-3). The ECM03-003 (MPR2390 R-3) calculation provides system capability clarification with regards to operation and LTOP tolerance that does not change the likelihood or consequences of a malfunction of a structure, system, or component important to safety described in the UFSAR, and does not change the conclusions reached utilizing ECM03-003 revision 0. The existing two train design of SDC permits placement of the alternate SDC train into service should the SI-405A(B) valve fail to operate because of a malfunction or an operator error. The ECM03-003 (MPR2390 R-3) calculation concludes that, should the SI-4052A(B) valve fail to open, SDC can be placed into service without the potential to lift the LTOP relief valve, by reducing RCS pressure to less than 230 psia and opening the SI-405A(B) valve. The saturation temperature for 230 psia is 394°F. To achieve the required 28°F subcooled margin for 230 psia and the temperature limit for SDC in OP-902-008, "Functional Recovery", T_H could be as high as 350°F, which is the normal SDC temperature limit. Therefore, reducing pressure to less than 230 psia would not require additional cooldown time to reach SDC entry conditions.

The addition of the fill bypass line decreases the likelihood of a pressure transient occurring when SI-405A(B) is opened. The SDC suction gas accumulation is not caused by this modification. EC-14765 / ECN25944 and Calculation ECM03-003 (MPR2390 R-3) only enhance the existing system and provide revised requirements to minimize the consequences of the already existing gas accumulation condition. Thus, there is not more than a minimal increase in the consequences of a malfunction of a structure, system, or component previously evaluated in the UFSAR.

The installation of the bypass system to the SDC system will not affect the system's previously evaluated malfunction consequence as part of the reactor coolant pressure boundary as the bypass system is designed to the same safety classifications as the SDC system. The bypass system is isolated from the RCS pressure when the RCS is above 350F and 392 psig. The RCS pressure isolation is provided by valve SI-401A(B) being closed during normal plant operation with the RCS above 350F and 392 psig. A potential malfunction of the new ¾" fill / pressurization line (failure of the bypass valve to actually open as expected) while placing SDC system into service could cause a pressure transient that could lift the LTOP, but would be isolatable using the SI-405A(B) valve and could not cause dose consequences beyond what has been analyzed in ECS04-001, "LOCA AST Radiological Dose Consequences", and described in the UFSAR accident analyses. This modification would not adversely affect the results of the existing, and bounding UFSAR LOCA analysis of the existing 14" shutdown cooling piping. The existing redundant trains of SDC provide alternate means of SDC should the bypass orifice become blocked, or if the SI-4052A(B) valve failed to open.

The ability to recover from inadvertent action to close the breaker for SI-405A(B) and SI-4052A(B) valves is similar to the recovery presently evaluated for SI-401A(B) valve breaker closure. The recovery time is enveloped in the recovery time for closure of the SI-401A(B) valve breakers. The ability to recover from a credible failure to hold SI-4052A(B) open for at least 10 minutes prior to opening SI-405A(B) is assured based on operator training regarding actions to

take in the event of an inadvertent LTOP relief valve actuation. Training demonstrates that operators can quickly recognized the loss of reactor coolant inventory through the SI-406B valve and respond to close SI-405A(B) before pressure level is depleted. The consequences would be bounded by the LOCA analysis. The recovery from a credible error in performance of the manual action evaluated here and the expected time to make such a recovery would not result in more than a minimal increase in consequences evaluated in the UFSAR.

Thus, this modification does not result in more than a minimal increase 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 types of accidents that could be caused by this modification are loss of coolant (due to piping failure) or containment boundary leakage (due to valve failure). These present UFSAR accident evaluations are considered not impacted by the change proposed.

The new configuration will also not affect shutdown cooling conditions. SI-4052A(B) has the same requirements as SI-405A(B) and any failure of SI-4052A(B) would be bounded by that already analyzed for SI-405A(B).

No new accident initiators or types are created by this change as the new piping system will meet the necessary code and design requirements and equivalent equipment interlocks and administrative controls are used.

The ECM03-003 calculation revision 1 clarifications do not change conditions of the existing small break and large break LOCA accidents evaluated in the UFSAR. The operational clarification provided by the ECM03-003 revision 1 calculation for gas voids in the LPSI SDC does not create additional possible accidents different from the small and large break LOCA in the UFSAR. The conclusions determined from ECM03-003 revision 0 calculation remain unchanged with regards to plant accidents for the changes made in revision 1 of the calculation.

The blockage failure of both bypass orifices in the SDC "A" and "B" trains [similar to the failure of both of the SI-4052A and SI-4052B valves to open] is shown analytically to not prevent putting SDC into service through use of bounding maximized gas voids with an RCS pressure as low as 70 psia while demonstrating the maximized gas void does not gas bind the LPSI pumps and as high as 230 psia while demonstrating that the LTOP relief valve will remain seated. The simultaneous failure of both bypass loops was analyzed in ECM03-003 revision 1 under LOCA conditions of a maximized gas void in the containment penetrations 40 and 41 including the SDC piping between SI-405A(B) and SI-407A(B) combined with gas voids downstream of the SI-407A(B) valves. The analysis concludes that the LPSI pumps wound not be gas bound, and would transport the maximized gas void within two minutes of being placed into operation. The reduced LPSI pump flow for this initial two minutes is within the operator time to energize the motive driver for SDC valves SI-405A(B) and SI-407A(B), and as such does not create the possibility of an accident of a different type than any previously evaluated in the existing UFSAR. The Waterford 3 plant administratively controls the size of the potential SDC gas void through OP-903-026 instructions that vent, or measure gas voids in the SDC piping downstream of SI-407A(B). To limit the accumulation of gas voids in the reactor containment penetrations 40 and 41, and SDC piping, the RCS leakage limit of 0.26 gallons per minute was determined by analysis as acceptable and is bounded by the existing OP-040-000 investigative action at 0.1 gpm for RCS unidentified leakage. Through the administrative control of monitoring

RCS leakage through valve SI-401A(B), the potential of gas accumulation in the reactor containment penetrations 40 and 41 and SDC piping is minimized to a LPSI pump tolerable condition for pump function. In addition, abnormal gas accumulation downstream of SI-407A(B) would be discovered, identified, and controlled through the condition report process, in accordance with the administrative controls of OP-903-026. No common failure mechanism is being caused, rather controls and limits will identify and preclude a possible "unknown" plant condition that would place both redundant SDC trains beyond the analyzed condition with regards to gas voiding.

Therefore, the proposed change does not create the possibility of 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 previously experienced Shutdown Cooling System pressure transient and relief valve lift (CR-WF3-2008-4161) ~~is~~was a result of a rapid compression of an accumulated gas void in the piping. The proposed bypass line addition will control the void compression and limit the transient pressure below the relief valve setpoint ensuring the system design and licensing basis is maintained. _A malfunction of the new bypass system solenoid valve or in the new piping or pipe supports could result in a loss of coolant or in containment boundary leakage and would lead to the same results as have been previously analyzed in the UFSAR. ~~The new configuration will also not affect shutdown cooling conditions. SI-4052A(B) has the same requirements as SI-405A(B) and any failure of SI-4052A(B) would be bounded by that already analyzed for SI-405A(B).~~ No malfunction of the added piping or valves could lead to a different result than has been previously evaluated.

The ECM03-003 calculation revision 1 clarifications do not change conditions to the existing small break and large break LOCA accidents evaluated in the UFSAR. The operational clarification provided by the ECM03-003 revision 1 calculation for gas voids in the LPSI SDC does not create additional possible malfunction of structure, system, or component (SSC) different from the small and large break LOCA previously evaluated in the UFSAR. The conclusions determined from ECM03-003 revision 0 calculation remain unchanged with regards to plant accidents for the changes made in revision 1 of the calculation.

The blockage failure of both bypass orifices in the SDC "A" and "B" trains [similar to the failure of both of the SI-4052A and SI-4052B valves to open] is shown analytically to not prevent putting SDC into service through use of bounding maximized gas voids with an RCS pressure as low as 70 psia while demonstrating the maximized gas void does not gas bind the LPSI pumps and as high as 230 psia while demonstrating that the LTOP relief valve will remain seated. The simultaneous failure of both bypass loops was analyzed in ECM03-003 revision 1 under LOCA conditions with a maximized gas void in the containment penetrations 40 and 41 including the SDC piping between SI-405A(B) and SI-407A(B) combined with gas voids downstream of the SI-407A(B) valves. The analysis concludes that the LPSI pumps would not be gas bound, and would transport the maximized gas void within two minutes of being placed into operation. The reduced LPSI pump flow for this initial two minutes is within the operator time to energize the motive driver for SDC valves SI-405A(B) and SI-407A(B), and as such does not create the possibility of a malfunction with a different result than any previously evaluated in the existing UFSAR. The Waterford 3 plant administratively controls the size of the potential SDC gas void

through OP-903-026 instructions that vent, or measure gas voids in the SDC piping downstream of SI-407A(B). To limit the accumulation of gas voids in the reactor containment penetrations 40 and 41, and SDC piping, the RCS leakage limit of 0.26 gallons per minute was determined by analysis as acceptable and is bounded by the existing OP-040-000 investigative action at 0.1 gpm for RCS unidentified leakage. Through the administrative control of monitoring RCS leakage through valve SI-401A(B), the potential of gas accumulation in the reactor containment penetrations 40 and 41 and SDC piping is minimized to a LPSI pump tolerable condition for pump function. In addition, abnormal gas accumulation downstream of SI-407A(B) would be discovered, identified, and controlled through the condition report process, in accordance with the administrative controls of OP-903-026. The operation of the SDC LPSI remains within the design parameters of the plant equipment and does not create a malfunction with different results than any previously evaluated in the UFSAR.

The ability to recover from inadvertent action to close the breaker for SI-405A(B) and SI-4052A(B) valves is similar to the recovery presently evaluated for SI-401A(B) valve breaker closure. The recovery time is enveloped in the recovery time for closure of the SI-401A(B) valve breakers. The ability to recover from a credible failure to hold SI-4052A(B) open for at least 10 minutes prior to opening SI-405A(B) is assured based on operator training regarding actions to take in the event of an inadvertent LTOP relief valve actuation. Training demonstrates that operators can quickly recognized the loss of reactor coolant inventory through the SI-406B valve and respond to close SI-405A(B) before pressure level is depleted. The consequences would be bounded by the LOCA analysis. The recovery from a credible error in performance of the manual action evaluated here and the expected time to make such a recovery would not result in more than a minimal increase in consequences evaluated in the U FSAR.

Therefore, the proposed change does not create 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.

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 impacts two fission product barriers. It adds piping and valves to the reactor coolant pressure boundary and it adds a new containment isolation valve. SI-4052A(B) has the same requirements as SI-405A(B) and any failure of SI-4052A(B) would be bounded by that already analyzed for SI-405A(B). Equivalent interlocks and administrative controls are used to control the operation of the new SI-4052A(B) valve and existing SI-405A(B) valve. Any accident or malfunction involving the new piping/valves will be bounded by existing analyses in the UFSAR and consequently design basis limits for fission product barriers will not be impacted.

Therefore, the proposed change 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:

All analyses performed to support this modification are consistent with existing UFSAR described methodologies. Considering the proposed change is limited in scope to the Reactor Coolant / Shutdown Cooling System, the previously approved methods used to evaluate reactor coolant pressure boundary were used without change. The ECM03-003 calculation revision 1 analysis methodology is not explicitly described in the UFSAR and did not change or result in a change to

those methods of evaluation described in the existing 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 analysis.

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 # 2014-01 / Rev. #:** 0

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.

[1] 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 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, battery loading, diesel loading, heat loading and fire safe shutdown.

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 revised to consider the addition of the Swing SUPS on diesel generator, HVAC, MCC and battery loading. 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: R. G. Finkenaur / See AS / DP Engineering / Design Engineering / Jon Langberg (Training) / / DP Engineering / Design Engineering /
Name (print) / Signature / Company / Department / Date

Reviewer: Evans Heacock / See AS/ DP Engineering / Design Engineering/
Name (print) / Signature / Company / Department / Date

OSRC: /
Chairman's Name (print) / Signature / Date
OSRC Meeting Number: W3 14-02

II. 50.59 EVALUATION

LICENSING BASIS

FSAR 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.

8.3.1 AC POWER SYSTEMS
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

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) During normal plant operation the SUPS provides 120 VAC power from the rectifier through the inverter. On loss of 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 can assume the load of the PDP. On SUPS A and B, transfer of PDP from inverter to bypass is automatic via the static switch which electronically switches the supply feed from inverter output to bypass output on loss of inverter. SUPS MA through MD have bypass capability but the transfer is manual, therefore, if a failure occurs resulting in loss of 120VAC, a panel outage will occur.

d) Action of the manual transfer of the 120V bus to the bypass transformer is annunciated in the main control room.

e) 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 b. Cable fault 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.
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.
3. Loss of 480VAC power to SUPS	a. MCC bus fault 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.

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.

Moving 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 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".

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. Loading on the diesel generators is not increased.

Calculation ECE89-005 "Switchgear Room 3B Heat Load at the Inception of SBO" evaluates the heat loading of replacement SUPS with the fourth SUPS unloaded but energized. The existing calculations

bound the case for normal operation during Phases I and II since the Swing SUPS functionally takes the place of the existing SUPS and the calculation uses the full rating of the SUPS and not actual loading.
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 Swing SUPS inverters as continuous loads to the DC system (Batteries and Chargers). This will allow the swing SUPS to be energized while unloaded, thus minimizing their start-up time should their use become necessary.
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.
Calculation ECE96-001, "Heat Released by Electrical Equipment in the RAB SWGR Area" is impacted by the last installation phase to show the swing SUPS additional heat load.
EE5-32-2 "Appendix R- Associated Circuit Analysis-Coordination Study" is revised to add new cables for the new PDPs installed to replace the internal PDPs in SUPS MA, MB, MC and MD and the associated transfer switches.
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); therefore, the consequences of any accident involving SUPS bus loss 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:

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.

ATTACHMENT 9.4 **NRC SUBMITTAL REVIEW**

Sheet 1 of 2 *{Typical}*

Letter #: W3F1-2014-0028

Response Due: 4/30/2014

Subject: Report of Facility Changes, Tests and Experiments and Commitment Changes for two year period starting April 26, 2012 and ending April 25, 2014.

Date Issued for Review: 4/30/2014

Correspondence Preparer / Phone #: Lillian Brown/ 504-739-7384

Section I **Letter Concurrence and Agreement to Perform Actions**

POSITION / NAME	Action (concurrence, certification, etc.)	Signature (sign, interoffice memo, e-mail, or telecom)
NA	NA	NA
COMMENTS		

Section II **Correspondence Screening**

Does this letter contain commitments? **If "yes," identify the commitments with due dates in the submittal and in Section III. When fleet letters contain commitments, a PCRS LO (e.g., LO-LAR, LO-WT) should be initiated with a CA assigned to each applicable site to enter the commitments into the site's commitment management system.**	Yes No	☐ ☒
Does this letter contain any information or analyses of new safety issues performed at NRC request or to satisfy a regulatory requirement? **If "yes," reflect requirement to update the UFSAR in Section III.**	Yes No	☐ ☒
Does this letter require any document changes (e.g., procedures, DBDs, FSAR, TS Bases, etc.), if approved? **If "yes," indicate in Section III an action for the responsible department to determine the affected documents. (The Correspondence Preparer may indicate the specific documents requiring revision, if known or may initiate an action for review.)**	Yes No	☐ ☒

Does this letter contain information certified accurate? **If "yes," identify the information and document certification in an attachment. (Attachment 9.5 must be used.)**	Yes No	☐ ☒

ATTACHMENT 9.4 **NRC SUBMITTAL REVIEW**

Sheet 2 of 2

Section III **Actions and Commitments**

Required Actions *Note: Actions needed upon approval should be captured in the appropriate action tracking system*	**Due Date**	**Responsible Dept.**
NA	NA	NA
Commitments *Note: When fleet letters contain commitments, a PCRS LO should be initiated with a CA assigned to each applicable site to enter the commitments into the site's commitment management system.*	**Due Date**	**Responsible Dept.**
NA	NA	NA

Section IV **Final Document Signoff for Submittal**

Correspondence Preparer	Lillian L. Brown
Final Submittal Review *(optional)*	Jim Pollock
Responsible Department Head	John Jarrell

ATTACHMENT 9.5 **CERTIFICATION REFERENCE FORM**

Sheet 1 of 1

{Typical}

Letter Number: **W3F1-2014-0028**

Subject: **Report of Facility Changes, Tests and Experiments and Commitment Changes**

Certifiable Statement(s): Use one of the following methods to identify certifiable statements in the table below:

1 Identify location in submittal (e.g., page 3, para 2, sentence 1) OR,
2 Paste in the exact words of the statement(s) OR,
3 State "see attachment" and attach a copy of the correspondence with the certifiable statements indicated (e.g., by redlining, highlighting, or underlining, etc.).

Each statement or section of information being certified should be uniquely numbered to correspond with the supporting documentation listed below.

Objective Evidence or Basis of Peer Review: List the supporting documents in the table below and attach a copy of the documents OR give basis of peer review. Large documents need not be attached.

Certifiable Statement(s)	Objective Evidence or Basis of Peer Review
1. This report provides a summary of changes made pursuant to 10CFR50.59 (d)(2) for the period from April 26, 2012 through April 25, 2014.	1. The 50.59 evaluation table was reviewed for all changes made through April 25, 2014 that had not yet been reported to the NRC in reports submitted pursuant to 10CFR50.59(d)(2). A review of eB, Indus Asset Management, was made for all of the identified changes for evidence of implementation. All changes not previously reported to the NRC pursuant to 10CFR50.59(d)(2) that have been implemented through April 25, 2014 are included in the report.
2. This report provides copies of the 50.59 submittals made pursuant to 10CFR 50.59(d)(2) for the period from April 26, 2012 through April 25, 2014	2. 12-01 EC-0000030976 R000, SI-1239A(B) Backup Air Supply; validated by review of EC status closeout complete. 12-02 CR-WF3-2012-3280, Documents Valve ACC-126A (CCW 'A' Temperature Control Valve) associated with procedure EP-002-100 Technical Support Center (TSC) Activation, Operations and Deactivation in order to close ACC-126A to

	prevent excess water inventory losses, validated closure by review of PCRS. 10-09-01 EC-0000008439 Miscellaneous Rupture Restraint modification for SG/RVCH replacement; validated by review of EC status as closeout complete. 12-03 ECN-0000040132 R000 (EC8458) Incorporation of RSG Design Document Updates and review of Design Basis Methodologies for RSG design report; validated by review of EC status as closeout complete. 12-04 EC-0000030663 R000 W3C19 Core Reload 50.59 Evaluation; validated by review of EC status as closeout complete. 12-05 EC-0000041095 R000 Backup Air Supply System to Nitrogen Accumulators 1 and 2; validated by review of EC status as closeout complete. 12-06 Impact of MSSV/ADV Leakage on Accident Dose Consequences; validated by review of EC status as closeout complete. 12-07 EC-0000025199 R000 GSI 191 (Generic Safety Issue) Margin Reduction for Safety Injection Sump; validated by review of EC status as closeout complete. 12-08 EC-0000041355 R000 Evaluate manual operator actions outside Control Room for certain Air Operated Valves after Accumulator is Exhausted; validated by review of EC status as closeout complete. 13-01 EC-0000043821 R000 Safety Injection Tank 2B, Nitrogen Supply Temperature; validated by review of EC status as closeout complete. 10-06-01 EC-0000014765 R000 SI-405A(B) bypass Fill/Equalization line addition/ECN-25944, changes for Calculation MPR-2390 R3 SDC Gas Intrusion Analysis; validated by review of EC status as closeout complete. 14-01 EC-0000043927 R000 Vital and

	Instrument SUPS Upgrade Project; validated by review of EC status as closeout complete.
3. This report also provides a summary of commitment changes made during that same period.	3. Commitment changes were extracted by Regulatory Assurance from CCEF database to identify all commitment changes that were approved through April 25, 2014. Licensing screen print of CCEF database and CCEF files provided electronically in J drive in submittal folder.

Individual certifying the statement(s): Certification may be documented using e-mail, telecom, "sign off" sheet, or inter-office memorandum. The form of documentation should specifically identify the information being certified.

Lillian L. Brown	LICENSING	4/28/2014
Name Lillian L. Brown	Department Licensing	Date 4/28/2014

Peer Review: Prior to signing for certification, determine if a Peer Review is required per section 5.4[2](c). Indicate "N/A" if not required.

Jim Pollock	Licensing	4/28/14
Name Jim Pollock	Department Licensing	Date 4/28/2014