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

W3F1-2016-0039

April 28, 2016

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

Subject: Report of Facility Changes, Tests and Experiments and Commitment Changes for two year period ending April 28, 2016
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 28, 2014 through April 28, 2016 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, Regulatory Assurance Manager 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
U.S. Nuclear Regulatory Commission
Region IV
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U.S. NRC Project Manager for Waterford 3

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Attachment

W3F1-2016-0039

Summary of Evaluations

10CFR50.59 Evaluation Number	Initiating Document	Summary
14-02	EP-002-100	Technical Support Center (TSC) Activation, Operation, and Deactivation procedure; timing of required operator action to replenish water in (CSP) Condensate Storage Pool & the (WCT) Wet Cooling Tower basins.
15-01	CR-WF3-2015-3565	Operability Compensatory Measure; manual action of EFW backup control valves.
15-01 R1	CR-WF3-2015-3565	Operability Compensatory Measure; manual action of EFW backup control valves. The revision improves the documentation and change the operator required action time to within 30 minutes.
15-02	ECN-0000047846 R000	LBDCR 15-019 (Licensing Basis Document Change Request) Fukushima Emergency Preparedness Communications.
15-03	EC-0000058901 R000	Evaluate Reactor Coolant System RC-IT0125-1 for use as CPC input (Core Protection Calculator.
15-04	EC-0000054054 R000	Base EC for WC321 – Cycle 21 Reload Process Applicability (PAD) identified 2 adverse changes. Both changes were demonstrated to be within the design requirements.
15-05	EC-0000061746 R000	Temporary modification for Control Element Drive Mechanism Control System Position switches. This activity does not affect the Core Protection Calculator (CPC) monitoring of target rod indications.
16-01	CR-WF3-2015-6448	LBDCR 16-006 will update FSAR section 7.7 (Pressurizer Level Control System) and section 9.3 (Chemical and Volume Control System) to resolve CR identified conflict and to make the FSAR sections consistent with each other.

10CFR72.48 Evaluation Number	Initiating Document	Summary
None		

Enclosure to W3F1-2016-0039

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

(104 pages)

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
2015-0002	P-27261	Security Monitoring & Assessment system Compensatory Measure 21 (Cancelled 3/19/15 Security)	Remove commitment; the commitment has been deemed unnecessary due to the modifications that have been installed and the changes implemented with the Protective Strategy have rendered this commitment obsolete.
2015-003	P-27243	Security Monitoring & Assessment computer system (Cancelled 3/19/2015 Security)	Remove commitment; the commitment has been deemed unnecessary due to the modifications that have been installed and the changes implemented with the Protective Strategy have rendered this commitment obsolete.
2015-004	P-6205	Materials Handling Program - Addressing dynamic loads in the selection of lifting devices not specially designed - Replaces MM-001-006 Sling Inspection and control. (Historic 2/4/15 Work Management)	Commitment changed to Historic due to a Corporate procedure replacement for this plant function. EN-MA-19
2015-005	P-9624	Materials Handling Program - Items to be checked out during frequent inspections of Cranes - Replaces MM-01-006 Sling Inspection and control. (Historic 2/4/15 Work Management)	Commitment changed to Historic due to a Corporate procedure replacement for this plant function. EN-MA-19
2015-006	P-12567	Operation of the 4 HPSI pump miniflow recirculation valves has been changed from automatic to manual operation. The guidelines for the operator stipulating when he must initiate closure of these valves upon actuation of RAS will be incorporated into the functional based EOPs. (Historic 5/18/15 Engineering) (Procedures referenced – OP-902-002 Loss of Coolant Accident Recovery, OP-902-008 Functional Recovery and ECS98-001 EOP Action Value Basis Document.	A RWSP isolation valve closure time was a requirement due to dose calculation methodology in Regulatory Guide 1.4 (TID-14844) but is no longer a requirement under Alternative Source Term (AST) dose methodology (Reg Guide 1.83). Current procedures and calculations (see references) will maintain the current two-minute requirement to close the recirculation valves for conservatism.

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
2015-008	P-8537	CE-001-001 Training for Chemistry technicians (Superseded Chemistry 3/2/15)	Summary of Justification for change – Site Procedure CE-001-001 is being deleted and is superseded by EN-CY-100
2015-009	P-8531	CE-001-001 Training for Chemistry technicians (Superseded Chemistry 3/2/15)	Summary of Justification for change – Site Procedure CE-001-001 is being deleted and is superseded by EN-CY-100
2015-010	P-279	EN-MA-118 Foreign Material Exclusion reference UNT-007-005 Housekeeping & Cleanliness – Contamination Prevention (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-05 is changing to Historic and is replaced by EN-MA-118.
2015-011	P-3151	In-service Testing of Snubbers - MM-007-011 (Historic Work Management 4/29/2015)	Summary of justification for change – Site Procedure MM-007-011 is changing to Historic due to CEP-SNB-0001 Dynamic Restraint Snubber exam & testing program
2015-012 – 2015-030	P-4675 P-9142 P-9143 P-9144 P-9145 P-9146 P-9147 P-9148 P-9154 P-9155 P-9156 P-9157 P-9158 P-9161 P-9180 P-9183 P-9184 P-9185 P-9187	Quality Instructions – Cleanliness Inspections UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-05 is changing to Historic and is replaced by EN-MA-118

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
2015-0031	P-9189	Installation of Cleaning of Fluid Systems & Components Protect Large Opening from Falling & Windblown Contaminants UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-05 is changing to Historic and is replaced by EN-MA-118
2015-0032	P-194	Admin Controls – Equipment Control – Extraneous Material Control UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-05 is changing to Historic and is replaced by EN-MA-118
2015-0033	P-5583	Inspection of Equipment and Rigging MM-001-006 (Historic Work Management 1/28/2015)	QAPM for Waterford 3 replaced by corporate QAPM and no longer tracks commitments.
2015-0034 – 2015-0063	P-11606 P-11607 P-11609 P-11610 P-11611 P-11612 P-11613 P-11615 P-11616 P-11617 P-11620 P-11621 P-11622 P-11623 P-11624 P-11625 P-11626 P-11628 P-11629 P-11630 P-11631 P-11632 P-11633 P-11634 P-11635 P-11636	Safety Standards for Slings, AlloySteel Chain type rated loads. MM-001-006 Sling Inspection and Control. (Historic Work Management 1/28/2015)	Summary of Justification for change – Site Procedure MM-001-006 is changing to Historic and is replaced by EN-MA-118

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
	P-11637 P-11638 P-11639 P-11936		
2015-0064	P-14526	Information concerning the use of anaerobic/adhesive sealants Procedures MM-006-001 Valve Maintenance MM-006-002 Valve Operator Maintenance MM-006-023 Type 1 mechanical snubber (shock arrestor maintenance, MM-007-001 BOP safety and relief valve bench testing and maintenance. (Historic Work Management 1/28/2015)	Summary of Justification for change – Site Procedure MM-001-006, MM-001-002, MM-006-23 are changing to Historic and are replaced by Corporate procedure EN-EV-112 Chemical Control Program.
2015-0065	P-15432	Degradation of threaded fasteners in the reactor coolant pressure boundary of power plants MM-006-001, MM-006-003 and MM-08-047 (Historic Work Management 1/28/2015)	Summary of Justification for change – Site Procedure MM-001-006, MM-006-003, MM-008-047 are changing to Historic and are replaced by Corporate procedure EN-MA-145 Maintenance Standard for Torque Applications
2015-0066	P-15433	Degradation of threaded fasteners in the reactor coolant pressure boundary of power plants MM-006-001, MM-08-047 (Historic Work Management 1/28/2015)	Summary of Justification for change – Site Procedure MM-001-006, and MM-008-047 are changing to Historic and are replaced by Corporate procedure EN-MA-145 Maintenance Standard for Torque Applications
2015-0067	P-329	Surveillance Requirements applicability – ISI and ASME testing MM-006-001Valve Maintenance and MM-007-004 Pressurizer Safety Valve Test (Historic Work Management 1/28/2015)	Summary of Justification for change – Site Procedure MM-006-001, and MM-007-004 are changing to Historic and are replaced by Corporate procedure CEP-RR-001, ASME section XI repair and replacement program CEP-IST-4 standard on in-service testing.
2015-0068	P-10298	Effects of oil on elastomeric materials used in ASCO NP-1 solenoid valves (Instrument Air System) MM-006-001 Valve Maintenance and MM-007-001 BOP safety and relief valve bench testing and maintenance.	Summary of Justification for change – Site Procedure MM-006-001, MM-007-001, are changing to Historic and are replaced by Corporate procedure EN-EV-112 Chemical

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
			Control Program
2015-0069	P-15435	Degradation of threaded fasteners in the reactor coolant pressure boundary of power plants MM-006-001, MM-006-019 (Historic Work Management 1/28/2015)	Summary of Justification for change – Site Procedure MM-006-001, MM-006-019, are changing to Historic and are replaced by Corporate procedure EN-MA-145 Maintenance Standard for Torque Applications
2015-0070	P-15436	Degradation of threaded fasteners in the reactor coolant pressure boundary of power plants MM-006-001,Valve Maintenance, MM-006-003 Rotated Equipment maintenance, MM-006-011 General Torquing and detention MM-008-047 Steam Generator primary services (Historic Work Management 1/28/2015)	Summary of Justification for change – Site Procedure MM-006-003, MM-006-011, MM-008-047 are changing to Historic and are replaced by Corporate procedure EN-MA-145 Maintenance Standard for Torque Applications
2015-0071	P-21390	IST Plan Pump & Valve R7C4 SI-107A, SI-107B Develop task for Disassembly and Inspection MM-006-052 Check Valve Inspection (Dual Plate) (Historic Work Management 10/29/2015)	MM-006-052 Check Valve Inspection (Dual Plate) is a Maintenance Procedure that Disassembles, Inspects and reassembles Dual Plate Check Valves
2015-0072	P-16588	This is an INPO Commitment not a regulatory commitment. Check Valve Failure or Degradation. MM-006-052 Check Valve Inspection (Dual Plate) (Historic Work Management 10/29/2015)	The SOER 86-03 (INPO) Check Valve Failures EPG Eng Program Guide Ck valve program
2015-0073	P-16590	This is an INPO Commitment not a regulatory commitment. Check Valve Failure or Degradation. MM-006-052 Check Valve Inspection (Dual Plate) (Historic Work Management 10/29/2015)	The SOER 86-03 (INPO) Check Valve Failures EPG Eng Program Guide Ck valve program
2015-0074	P-9190	Cleaning of Fluid System Control of Access into System after cleaning UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118
2015-0075	P-9191	Cleaning of Fluid Seals to be installed to Protect against Environmental Contamination & Quality Degradation UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
2015-0076	P-9192	Precautions necessary to Preserve cleanness of Sealed Fluid System when access is required. UNT-007-005 Cleanliness control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118
2015-0077	P-9193	Preop Cleaning of Fluid System Isolated areas where cleaning is performed to minimize Interference UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118
2015-0078	P-9194	Preop Cleaning of Fluid System Personnel to be Familiar with Procedure and Associated Hazards UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118
2015-0079	P-9196	Preop Cleaning of Fluid System Loose Tools to be Attached to workman or exterior of system with lanyard. UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118
2015-0080	P-9199	Preop Cleaning of Fluid System Interior of Accessible Components to be inspected for Debris Contaminants UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118
2015-0081	P-9215	Cleaning of Fluid System Chelate Cleaning UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118
2015-0082	P-9216	Fluid System Cleaning precautions Chloride Stress cracking inhibitor for systems containing Austenitic Stainless Steel UNT-007-005 Cleanliness Control	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
		(Historic Work Management 4/29/2015)	
2015-0083	P-9218	Fluid System Cleaning Precautions Use of Acid Chelating agents on certain materials in not recommended. UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005,is changing to Historic and is replaced by Corporate procedure EN-EV-112 Chemical Control Program.
2015-0084	P-9219	Fluid System Cleaning precautions use of Halogenated organic solvents UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005,is changing to Historic and is replaced by Corporate procedure EN-EV-112 Chemical Control Program.
2015-0085	P-9220	Fluid System Cleaning Precautions Acid cleaning of installed systems UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005,is changing to Historic and is replaced by Corporate procedure EN-EV-112 Chemical Control Program.
2015-0086	P-9221	Cleaning of Fluid Systems Control of Cleaning Solutions UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005,is changing to Historic and is replaced by Corporate procedure EN-EV-112 Chemical Control Program.
2015-0087	P-9224	Cleaning of Fluid Systems, Preparation, Collection, Storage and Maintenance of Records UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005,is changing to Historic and is replaced by Corporate procedure EN-EV-112 Chemical Control Program.
2015-0088	P-11153	QA requirements for installation, inspection and testing of Mech EQPT and System cleaning UNT-007-005 Cleanliness Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by Corporate procedure EN-EV-112 Chemical Control Program.
2015-0089	P-21722	Violation Response UNT-007-005 Revised to include responsibilities for Establishing and maintaining adequate cleanliness controls UNT-007-005 Cleanliness Control	Summary of Justification for change – Site Procedure UNT-007-005,is changing to Historic and is replaced by Corporate procedure EN-EV-112 Chemical Control

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
		(Historic Work Management 4/29/2015)	Program.
2015-0090	P-21911	Violation response Procedures revised to address RE&P responsibilities for Fuel Receipt and control of foreign material in SFP area UNT-007-005 Cleanliness Control. (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure UNT-007-005 is changing to Historic and is replaced by EN-MA-118
2015-0091	P-11614	Safety Standards for slings, alloy steel chain type inspection nicks and gouges MM-001-006 Sling Inspection and Control (Historic Work Management 4/29/2015)	Summary of Justification for change – Site Procedure MM-001-006 is changing to Historic and is replaced by EN-MA-119
2016-0001	A-27323	Basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection.	No change to commitment – commitment is being cancelled as it is not a regulatory commitment and is not required to be tracked
2016-0002	P-22575	Pending EC modification	
2016-0003	P-24776	Pending EC modification	

I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # 2014-02 / Rev. # 0**

Proposed Change / Document:
EP-002-100, "Technical Support Center Activation, Operation, and Deactivation"

Description of Change:
The timing of required operator action to replenish water in the Condensate Storage Pool (CSP) and the Wet Cooling Tower (WCT) basins following a design basis tornado is shortened in order to add operational margin to accommodate potential, but unexpected, gross leakage (11.5 gpm) past the seats of the Dry Cooling Tower (DCT) tube bundle isolation valves. The shortened time allowance is an adverse effect on the method of controlling a design function as described in FSAR section 9.2.5.3.3, "Site Related Phenomena".

The purpose of this 10CFR50.59 Evaluation is not to justify or evaluate any potential degraded condition (potentially leaking valves), but only to evaluate the effects of the proposed changes to add operational margin on other aspects of the facility or procedures described in the UFSAR (reference NEI 96-07 section 4.4). The attached Process Applicability Determination (PAD) provides additional details of the proposed change.

This 10CFR50.59 Safety Evaluation is required to determine if the change may be implemented without prior NRC approval.

Summary of Evaluation:
The 10CFR50.59 Evaluation concludes that the change may be implemented without prior NRC approval. Shortening the time allowed for replenishing the CSP and WCT basins does not involve any unreviewed safety question.

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 / [signature] / SMI / Projects / 11-1-2014
Name (print) / Signature / Company / Department / Date

Reviewer: Peter J. McKenna / [signature] / EOI / Systems Engineering / 11-1-2014
Name (print) / Signature / Company / Department / Date

OSRC: Carl Rich / SEE ATTACHED EMAIL / 11-1-2014
Chairman's Name (print) / Signature / Date

W3 14-16
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.

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 EVENT IMPACTED BY POTENTIAL, BUT UNEXPECTED, LEAKAGE OF THE DCT TUBE BUNDLE ISOLATION VALVES IS THE DESIGN BASIS TORNADO EVENT. CCW LEAKAGE IS NOT THE INITIATOR OF ANY ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR. THE REPLENISHMENT OF WATER INVENTORY IS NOT THE INITIATOR OF ANY ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR. THE PROPOSED CHANGES WILL ENSURE THAT ADEQUATE WATER INVENTORY FOR COOLING IS MAINTAINED FOR THE ENTIRE DESIGN BASIS TORNADO EVENT. THEREFORE, THE PROPOSED CHANGE DOES NOT AFFECT ANY ACCIDENT INITIATORS.

   THEREFORE, THE PROPOSED CHANGES SHORTENING THE TIME ALLOWED FOR REPLENISHING THE INVENTORY OF THE CSP AND WCT BASINS 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 ACCOMMODATES POTENTIAL, BUT UNEXPECTED, LEAKAGE PAST THE SEATS OF THE DCT TUBE BUNDLE ISOLATION VALVES. THIS EVALUATION DOES NOT ADDRESS ANY POTENTIALLY DEGRADED ISOLATION VALVES, BUT ONLY THE CHANGES THAT WOULD ACCOMMODATE SUCH LEAKAGE AS PER THE GUIDANCE IN NEI 96-07 SECTION 4.4. THE PROPOSED CHANGES WILL ENSURE THAT ADEQUATE WATER INVENTORY FOR COOLING IS MAINTAINED FOR THE ENTIRE DESIGN BASIS TORNADO EVENT. INVENTORY REPLENISHMENT IS ALREADY CREDITED FOR THE DESIGN BASIN TORNADO EVENT. THE PROPOSED CHANGE SHORTENS THE REQUIRED RESPONSE TIME TO COMMENCE REPLENISHMENT.

   THE RELEVANT SSCS IMPACTED BY THE DESIGN BASIS TORNADO EVENT ASSUMING POTENTIAL, BUT UNEXPECTED, ADDITIONAL LEAKAGE FROM THE CCW SYSTEM ARE THOSE SSCS REQUIRED FOR REPLENISHING WATER INVENTORY INTO THE CCW SURGE TANK, THE CSP, AND WCT BASINS IN ORDER TO MAINTAIN COOLING. NO ADDITIONAL SSCS ARE REQUIRED THAT WERE NOT PREVIOUSLY CREDITED AND NO ADDITIONAL MANIPULATION IS PERFORMED ON ANY PARTICULAR SSC THAN WAS CREDITED IN THE CURRENT ANALYSIS (MNQ9-17 AND ECM07-002). EC-53675 CONFIRMS THAT THE CCW MAKEUP PUMP IS CAPABLE OF PROVIDING THE ADDITIONAL FLOW TO INCLUDE THE POTENTIAL, BUT UNEXPECTED, LEAKAGE. THE SSCS INVOLVED WERE ALREADY REQUIRED BY THE ORIGINAL ANALYSIS AND ARE ONLY REQUIRED SOONER BECAUSE OF THE POTENTIAL, BUT UNEXPECTED, LEAKAGE. THEREFORE, THERE IS NO ADDITIONAL LIKELIHOOD OF AN EQUIPMENT MALFUNCTION AS A RESULT OF THE PROPOSED CHANGE.

   THE PROPOSED CHANGES IN THE TIME ALLOWED FOR REPLENISHING THE WATER INVENTORY IN THE CSP AND WCT BASINS SHORTENS THE ALLOWED RESPONSE TIME FOR CERTAIN OPERATOR ACTIONS. EACH OF THE OPERATOR ACTIONS AND THE ASSOCIATED MANIPULATED SSCS IS UNCHANGED EXCEPT FOR THE TIME ALLOWED FOR OPERATOR RESPONSE AFTER THE TORNADO EVENT. THE SAME SSCS ARE MANIPULATED IN THE SAME MANNER. THE SAME PROCEDURES WILL APPLY.

   THE FIRST OPERATOR ACTION AFFECTED IS REPLENISHMENT OF THE CSP. AS DETERMINED IN EC-53675, THE POTENTIAL, BUT UNEXPECTED, LEAKAGE THROUGH THE DCT TUBE BUNDLE ISOLATION VALVES CAUSES THE INVENTORY IN THE CSP TO BE DEPLETED IN APPROXIMATELY 3.71 HOURS RATHER THAN 3.92 HOURS. THE REQUIRED OPERATOR ACTIONS ARE DESCRIBED IN OP-902-003, "LOSS OF OFFSITE POWER / LOSS OF FORCED CIRCULATION RECOVERY", OP-902-006, "LOSS OF NORMAL FEEDWATER RESTORATION", OP-902-008, "FUNCTIONAL RECOVERY", AND OP-902-009, "STANDARD APPENDICES". OP-902-006 AND OP-902-008 REQUIRE GOING TO OP-902-009 WHEN THE CSP IS AT 25%. OP-902-009 REQUIRES REFILLING THE CSP BEFORE CSP REACHES 11%. OP-902-009 PROVIDES DETAILED INSTRUCTIONS FOR REFILLING THE CSP FROM THE WCT BASINS. MANUAL HANDWHEEL OPERATED VALVES ACC-115A(B), ACC-114A(B) AND ACC-116A(B) ARE MANIPULATED IN THE SAME MANNER. 3.7 HOURS IS AMPLE TIME TO COMPLETE THE OPERATION. 3.7 HOURS IS

ONLY SLIGHTLY LESS THAN 3.9 HOURS, WHICH IS CURRENTLY REQUIRED BY THE DESIGN BASIS CALCULATIONS. THE TIMING MEETS THE ANSI/ANS 58.8 CRITERIA OF 30 MINUTES FOR ACTIONS OUTSIDE THE CONTROL ROOM. THE APPLICABLE PROCEDURES DO NOT SPECIFY ANY PARTICULAR TIME AND THE ACTION IS IN RESPONSE TO INDICATIONS THAT ARE BEING MONITORED IN ACCORDANCE WITH PROCEDURES APPLICABLE TO THE EVENT. THE SAME INDICATIONS AND LIMITS ARE USED TO PROMPT OPERATOR ACTIONS. THEREFORE, NO CHANGES TO OP-902-003, OP-902-006, OP-902-008, OR OP-902-009 ARE REQUIRED TO ENSURE THAT THE OPERATOR ACTION IS PERFORMED CORRECTLY AND IN TIME TO MAINTAIN REQUIRED COOLING.

THE SECOND OPERATOR ACTION AFFECTED IS CROSS-CONNECTING THE WCT BASINS. AS DETERMINED IN EC-53675, THE POTENTIAL, BUT UNEXPECTED, LEAKAGE THROUGH THE DCT TUBE BUNDLE ISOLATION VALVES CAUSES THE INVENTORY IN ONE WCT BASIN TO BE DEPLETED IN APPROXIMATELY 10 HOURS, RATHER THAN 16 HOURS AS WAS EVALUATED WITHOUT LEAKAGE. THE REQUIRED OPERATOR ACTIONS ARE DESCRIBED IN EP-002-100, "TECHNICAL SUPPORT CENTER ACTIVATION, OPERATION, AND DEACTIVATION". EP-002-100, ATTACHMENT 7.3 DESCRIBES THE ACTION, WHICH INVOLVES OPENING REMOTE OPERATED VALVES ACC-138A AND ACC-138B. 10 HOURS IS AMPLE TIME TO OPERATE THE VALVE FROM EITHER THE CONTROL ROOM OR THE LOCAL HANDWHEEL AND PERFORM THE OPERATION. THE REQUIRED RESPONSE TIME WILL BE REVISED IN EP-002-100.

THE THIRD OPERATOR ACTION AFFECTED IS REPLENISHMENT OF THE WCT BASINS. AS DETERMINED IN EC-53675, THE POTENTIAL, BUT UNEXPECTED, LEAKAGE THROUGH THE DCT TUBE BUNDLE ISOLATION VALVES CAUSES THE INVENTORY IN THE WCT BASINS TO BE DEPLETED IN APPROXIMATELY 24 HOURS RATHER THAN 32 HOURS. THE REQUIRED OPERATOR ACTIONS ARE DESCRIBED IN EP-002-100, "TECHNICAL SUPPORT CENTER ACTIVATION, OPERATION, AND DEACTIVATION". EP-002-100, ATTACHMENT 7.3 DESCRIBES THE ACTION, WHICH INVOLVES OPENING HANDWHEEL OPERATED VALVE CW-402 AND THROTTLING OPEN CW-403A(B). VALVE CW-402 IS LOCATED IN A PIT UNDER THE TOOLROOM. A MANHOLE COVER MUST BE REMOVED TO ACCESS THE VALVE. 24 HOURS IS AMPLE TIME TO ACCESS THE VALVE AND PERFORM THE OPERATION. THE REQUIRED RESPONSE TIME WILL BE REVISED IN EP-002-100.

THE FOURTH POTENTIAL OPERATOR ACTION AFFECTED IS REPLENISHMENT OF THE WATER IN THE CIRCULATING WATER SYSTEM PIPING. AS DETERMINED IN EC-53675, THE POTENTIAL, BUT UNEXPECTED, LEAKAGE THROUGH THE DCT TUBE BUNDLE ISOLATION VALVES CAUSES THE INVENTORY IN THE CIRCULATING WATER SYSTEM PIPING TO BE DEPLETED IN APPROXIMATELY 45 HOURS RATHER THAN 68 HOURS. THE MANUAL ACTION INVOLVES SETTING UP A PORTABLE PUMP AND PUMPING WATER FROM THE MISSISSIPPI RIVER TO A HOLE CUT IN THE CIRCULATING WATER PIPING ON THE PLANT SIDE OF THE LEVEE. WHILE THIS IS A RATHER COMPLEX OPERATION, 45 HOURS IS STILL AMPLE TIME TO ESTABLISH WATER INVENTORY REPLENISHMENT TO MAINTAIN COOLING. THE IMPACTED PROCEDURE, EP-002-100, "TECHNICAL SUPPORT CENTER ACTIVATION, OPERATION, AND DEACTIVATION", WILL BE REVISED TO ENSURE THAT INSTRUCTIONS WITH THE NEW RESPONSE TIME REQUIREMENTS ARE AVAILABLE WHEN NEEDED.

TO ENSURE THAT THE PROPOSED SHORTENING OF RESPONSE TIMES DOES NOT CREATE A SITUATION IN WHICH OPERATORS DO NOT HAVE SUFFICIENT TIME TO CORRECTLY COMPLETE THE REQUIRED ACTIONS, THE NEW TIMING OF THE OPERATOR ACTIONS WAS EVALUATED AGAINST NRC REGULATORY ISSUE SUMMARY 2005-20, NRC INFORMATION NOTICE 97-78, AND ANSI/ANS-58.8-1994. RIS 2005-20 PROVIDES GUIDANCE REGARDING SUBSTITUTING MANUAL OPERATOR ACTIONS FOR AUTOMATIC ACTIONS. THE PROPOSED CHANGE DOES NOT SUBSTITUTE MANUAL OPERATOR ACTION FOR ANY AUTOMATIC ACTIONS. THE PROPOSED CHANGES INVOLVE ONLY CHANGES TO RESPONSE TIMES FOR EXISTING CREDITED MANUAL OPERATOR ACTIONS. NRC INFORMATION NOTICE 97-78, "CREDITING OF OPERATOR ACTIONS IN PLACE OF AUTOMATIC ACTIONS AND MODIFICATIONS OF OPERATOR ACTIONS, INCLUDING RESPONSE TIMES", PROVIDES GUIDANCE EMPHASIZING THE EXPECTATIONS TO PERFORM ADEQUATE SAFETY EVALUATIONS WHEN OPERATOR ACTION RESPONSE TIMES ARE SHORTENED. IN 97-78 ENDORSES THE USE OF ANSI 58.8, "TIME RESPONSE DESIGN CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS", WHICH STATES: "SAFETY-RELATED OPERATOR ACTIONS THAT ARE REQUIRED TO BE PERFORMED AFTER 30 MINUTES FOLLOWING THE INDICATION OF THE DESIGN BASIS EVENT MAY BE PERFORMED EITHER IN THE CONTROL ROOM OR AT LOCATIONS OUTSIDE THE CONTROL ROOM. IF THESE ACTIONS ARE PERFORMED OUTSIDE THE CONTROL ROOM, THE DESIGN MUST PROTECT THE OPERATOR AT LOCATIONS OUTSIDE THE CONTROL ROOM (INCLUDING ACCESS ROUTES THERETO) BY PROVIDING: (1) ADEQUATE PROTECTION FOR THE OPERATOR FROM THE ENVIRONMENTAL CONDITIONS CAUSED BY THE DBE FOR THE REQUIRED ACCESS TIME, ACTION TIME, AND RETURN TIME; (2) COMMUNICATIONS LINKS TO THE CONTROL ROOM; (3) EMERGENCY LIGHTING; (4) APPROPRIATE PROCEDURES FOR ACCOMPLISHING THE SAFETY-RELATED OPERATOR ACTION; AND (5) ALL NECESSARY TOOLS OR

SPECIAL EQUIPMENT FOR COMPLETING THE SAFETY-RELATED OPERATOR ACTION."

ADEQUATE PROTECTION FOR THE OPERATOR IS MAINTAINED BECAUSE THE ACTIONS REQUIRED IN LESS THAN 24 HOURS ARE PERFORMED WITHIN THE REACTOR AUXILIARY BUILDING, WHICH IS PROTECTED FROM TORNADO RELATED HAZARDS. ACTIONS PERFORMED OUTSIDE THE PROTECTIVE STRUCTURES ARE NOT REQUIRED FOR AT LEAST 24 HOURS, WHICH IS AMPLE TIME FOR THE SEVERE WEATHER TO PASS. THE PROPOSED CHANGE HAS NO IMPACT ON COMMUNICATION LINKS TO THE CONTROL ROOM OR EMERGENCY LIGHTING. THE PROCEDURES FOR THE PROPOSED ACTIONS ARE ALREADY ESTABLISHED AND THE ONLY CHANGES ARE SHORTENING RESPONSE TIMES. THE TOOLS AND SPECIAL EQUIPMENT ARE NOT CHANGED BY THE PROPOSED CHANGES. THE POST MODIFICATION TESTING FOR EC-530 (WO 108105) DEMONSTRATED THAT THE REPLENISHMENT ACTIVITIES INCLUDING SETTING UP THE PORTABLE PUMP NEAR THE MISSISSIPPI RIVER COULD BE COMPLETED IN TWO HOURS FROM THE TIME THAT THE ACTIVITY WAS COMMENCED.

THE GUIDANCE IN ANSI/ANS 58.8 IS MET BY THE NEW RESPONSE TIMES INDICATING THAT IT IS POSSIBLE TO CORRECTLY PERFORM THE ACTIONS WITHOUT CAUSING A MALFUNCTION. IN ADDITION, OPERATIONS AND EMERGENCY PLANNING PERSONNEL REVIEWED THE PROPOSED CHANGES AND CONCURRED THAT THE NEW TIMING IS ACHIEVABLE WITHOUT THE POSSIBILITY OF CREDIBLE ERRORS THAT COULD AFFECT THE OVERALL SAFETY FUNCTION.

THEREFORE, THE PROPOSED CHANGES SHORTENING THE TIME ALLOWED FOR REPLENISHING THE INVENTORY OF THE CSP AND WCT BASINS 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 RELEVANT EVENT IMPACTED BY THE POTENTIAL, BUT UNEXPECTED, LEAKAGE OF THE DCT TUBE BUNDLE ISOLATION VALVES IS THE DESIGN BASIS TORNADO EVENT. THE PROPOSED CHANGES WILL ENSURE THAT ADEQUATE WATER INVENTORY FOR COOLING IS MAINTAINED FOR THE ENTIRE DESIGN BASIS TORNADO EVENT. THEREFORE, THE PROPOSED CHANGE HAS NO EFFECT ON RADIATION DOSE. WATER REPLENISHMENT IS NOT NECESSARY OR INTENDED TO BE IMPLEMENTED FOR ANY OTHER ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR.

THEREFORE, THE PROPOSED CHANGES SHORTENING THE TIME ALLOWED FOR REPLENISHING THE INVENTORY OF THE CSP AND WCT BASINS 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 RELEVANT SSCS IMPACTED BY THE DESIGN BASIS TORNADO EVENT ASSUMING ADDITIONAL, BUT UNEXPECTED, LEAKAGE FROM THE CCW SYSTEM ARE THOSE SSCS REQUIRED FOR REPLENISHING WATER INVENTORY INTO THE CSP AND WCT BASINS IN ORDER TO MAINTAIN COOLING. NO ADDITIONAL SSCS ARE REQUIRED THAT WERE NOT PREVIOUSLY CREDITED AND NO GREATER RELIANCE IS PLACED ON ANY PARTICULAR SSC THAN WAS CREDITED IN THE CURRENT ANALYSIS. THE SSCS INVOLVED WERE ALREADY REQUIRED BY THE ORIGINAL ANALYSIS AND ARE ONLY REQUIRED SOONER BECAUSE OF THE POTENTIAL, BUT UNEXPECTED LEAKAGE. THIS EVALUATION DOES NOT ADDRESS ANY POTENTIALLY DEGRADED CONDITION, BUT ONLY THE EFFECTS OF THE CHANGES TO ADD OPERATIONAL MARGIN ON OTHER ASPECTS OF THE FACILITY AS PER THE GUIDANCE IN NEI 96-07 SECTION 4.4. THE PROPOSED CHANGES WILL ENSURE THAT ADEQUATE WATER INVENTORY FOR COOLING IS MAINTAINED FOR THE ENTIRE DESIGN BASIS TORNADO EVENT. INVENTORY REPLENISHMENT IS ALREADY CREDITED FOR THE DESIGN BASIN TORNADO EVENT. THE AREAS OF THE PLANT WHERE THE RELEVANT SSCS ARE LOCATED ARE NOT EXPECTED TO HAVE ANY SIGNIFICANT RADIATION DOSE RATES AT ANY TIME AFTER A DESIGN BASIS TORNADO EVENT. THEREFORE, THE PROPOSED CHANGE HAS NO EFFECT ON RADIATION DOSE. THE PROPOSED CHANGES ARE NOT NECESSARY OR INTENDED TO BE IMPLEMENTED FOR ANY OTHER MALFUNCTION PREVIOUSLY EVALUATED IN THE UFSAR.

THEREFORE, THE PROPOSED CHANGES SHORTENING THE TIME ALLOWED FOR REPLENISHING THE INVENTORY OF THE CSP AND WCT BASINS 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 RELEVANT TYPE OF ACCIDENT ASSOCIATED WITH THE PROPOSED CHANGE IS THE DESIGN BASIS TORNADO EVENT. THE PROPOSED CHANGE IS NOT APPLICABLE TO OTHER TYPES OF ACCIDENTS BECAUSE THE OTHER ACCIDENTS DO NOT CREDIT THE DCT TUBE BUNDLE ISOLATION VALVES. THE BASIS FOR THE RESPONSE TO QUESTION 2 CONCLUDES PROPOSED CHANGE DOES NOT RESULT IN MORE THAN A MINIMAL INCREASE IN THE LIKELIHOOD OF OCCURRENCE OF A MALFUNCTION. THEREFORE, THE PROPOSED CHANGE CANNOT CREATE THE POSSIBILITY OF ANY OTHER TYPE OF ACCIDENT.

THEREFORE, THE PROPOSED CHANGES SHORTENING THE TIME ALLOWED FOR REPLENISHING THE INVENTORY OF THE CSP AND WCT BASINS DOES NOT CREATE A 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 RELEVANT SSCS IMPACTED BY THE DESIGN BASIS TORNADO EVENT ASSUMING POTENTIAL, BUT UNEXPECTED ADDITIONAL LEAKAGE FROM THE CCW SYSTEM ARE THOSE SSCS REQUIRED FOR REPLENISHING WATER INVENTORY INTO THE CSP AND WCT BASINS IN ORDER TO MAINTAIN COOLING. NO ADDITIONAL SSCS ARE REQUIRED THAT WERE NOT PREVIOUSLY CREDITED AND NO GREATER RELIANCE IS PLACED ON ANY PARTICULAR SSCS THAN WAS CREDITED IN THE CURRENT ANALYSIS. THE SSCS INVOLVED WERE ALREADY REQUIRED BY THE ORIGINAL ANALYSIS AND ARE ONLY REQUIRED SOONER BECAUSE OF THE POTENTIAL, BUT UNEXPECTED, LEAKAGE. EC-53675 CONFIRMS THAT THE CCW MAKEUP PUMP IS CAPABLE OF PROVIDING THE ADDITIONAL FLOW TO COMPENSATE FOR THE POTENTIAL, BUT UNEXPECTED, LEAKAGE. THIS EVALUATION DOES NOT ADDRESS ANY POTENTIALLY DEGRADED CONDITION, BUT ONLY THE EFFECTS OF THE CHANGES TO ADD OPERATIONAL MARGIN ON OTHER ASPECTS OF THE FACILITY AS PER THE GUIDANCE IN NEI 96-07 SECTION 4.4. THE PROPOSED CHANGES WILL ENSURE THAT ADEQUATE WATER INVENTORY FOR COOLING IS MAINTAINED FOR THE ENTIRE DESIGN BASIS TORNADO EVENT. INVENTORY REPLENISHMENT IS ALREADY CREDITED FOR THE DESIGN BASIN TORNADO EVENT. THE AREAS OF THE PLANT WHERE THE RELEVANT SSCS ARE LOCATED ARE NOT EXPECTED TO HAVE ANY SIGNIFICANT RADIATION DOSE RATES AT ANY TIME AFTER A DESIGN BASIS TORNADO EVENT. THEREFORE, THE PROPOSED CHANGE DOES NOT CREATE ANY ADDITIONAL POSSIBILITY OF A MALFUNCTION AND HAS NO EFFECT ON RADIATION DOSE. THE PROPOSED CHANGES ARE NOT NECESSARY OR INTENDED TO BE IMPLEMENTED FOR ANY OTHER MALFUNCTION PREVIOUSLY EVALUATED IN THE UFSAR.

THEREFORE, THE PROPOSED CHANGES SHORTENING THE TIME ALLOWED FOR REPLENISHING THE INVENTORY OF THE CSP AND WCT BASINS 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 PROPOSED CHANGE DOES NOT DIRECTLY IMPACT ANY DESIGN BASIS LIMITS FOR ANY FISSION PRODUCT BARRIERS. THE COMPENSATORY MEASURES ENSURE THAT ADEQUATE COOLING IS MAINTAINED THROUGHOUT A DESIGN BASIS TORNADO EVENT, WHICH IS THE ONLY EVENT AFFECTED BY POTENTIAL LEAKAGE PAST THE DCT TUBE BUNDLE ISOLATION VALVES. THE PROPOSED CHANGE DOES NOT ADVERSELY IMPACT THE ABILITY OF THE EMERGENCY FEEDWATER SYSTEM OR THE ULTIMATE HEAT SINK TO PROVIDE HEAT REMOVAL AS ORIGINALLY ANALYZED. THE BASIS FOR THE RESPONSE TO QUESTION 2 CONCLUDES PROPOSED CHANGE DOES NOT RESULT IN MORE THAN A MINIMAL INCREASE IN THE LIKELIHOOD OF OCCURRENCE OF A MALFUNCTION. THEREFORE, THERE IS NO ADDITIONAL POSSIBILITY OF EXCEEDING FUEL CLADDING TEMPERATURE LIMITS.

THEREFORE, THE PROPOSED CHANGES SHORTENING THE TIME ALLOWED FOR REPLENISHING THE INVENTORY OF THE CSP AND WCT BASINS 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 EVALUATION TO DEMONSTRATE THE INTENDED DESIGN FUNCTION (MAINTAINING ADEQUATE WATER INVENTORY FOR COOLING AFTER A DESIGN BASIS TORNADO) WHILE ACCOMMODATING POTENTIAL, BUT UNEXPECTED, LEAKAGE PAST THE SEATS OF THE DCT TUBE BUNDLE ISOLATION VALVES IS PERFORMED USING THE SAME METHODS EMPLOYED IN THE EXISTING ANALYSIS.

THEREFORE, THE PROPOSED CHANGES SHORTENING THE TIME ALLOWED FOR REPLENISHING THE INVENTORY OF THE CSP AND WCT BASINS 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]

RJK 6/30/15

Facility: Waterford 3 **Evaluation # / Rev. #:** 15-01

Proposed Change / Document: CR-WF3-2015-3565, Manual Action for EFW backup flow control valves remaining in MANUAL MODE.

Description of Change: , OP-009-003 Steps 6.1.8 and 8.1.7 will be deviated (with component deviations) to allow Emergency Feedwater (EFW) Backup flow control valves (EFW-223A (B)) controllers to be placed in manual, which is not the normal configuration for this system.

Summary of Evaluation:
This 50.59 evaluation addresses the potentially adverse impact of the operator action required due to OP-009-003 Steps 6.1.8 and 8.1.7 being deviated to allow Emergency Feedwater (EFW) Backup Flow Control Valves (EFW-223A(B)) controllers to be placed in manual. This deviation is due to a condition identified in CR-WF3-2015-3565 which documented that EFW-223 A and B cycled during automatic flow control mode more than what is assumed in the nitrogen accumulator sizing analysis. With a loss of instrument air, the additional valve strokes could exhaust the safety related nitrogen accumulators prior to the credited 10 hours of remote valve operation.

Safety Function of EFW system is:

1. To provide sufficient supply of cooling water to the steam generators (SG) for the removal of decay heat from the reactor during emergency situations when the Main Feedwater System is not available, and
2. To maintain the SG level above the top of the SG tubes for mitigation of radiological dose consequences to the control room operators and general public.

Basic Operation of the Emergency Feedwater System
And normal operation of the flow control valves:

The EFW flow path consists of four flow control valves and four flow isolation valves. Each of the four flow control valves are operated independently from unique wide range level transmitters. EFW flow rates into the steam generators are controlled based on steam generator wide range levels.

EFAS is initiated when SG level decreases to 27.4% narrow range (72.5% wide range). No flow is initiated at this point, but isolation valves are opened, and the pumps are started.

As steam generator level decreases to 55% wide range (critical level) the primary flow control valve (FCV) opens to a fixed predetermined position to provide about 200 gpm (total flow). Actual flow will vary based on system conditions. If the primary valve does not supply sufficient flow, the backup FCV will open as required to provide about 175 gpm total flow.

If the SG level were to reach 45% wide range, the primary FCV maintains the fixed predetermined position equivalent to 200 gpm. The backup FCV will open as required to provide a total flow of 400 gpm to each steam generator.

At 36.3% wide range, a priority open signal is generated to fully open the primary and backup FCV to supply maximum flow to each steam generator.

The priority open signal is reset by 40% wide range level. The primary FCV returns to its fixed predetermined position (approximately 200 gpm). The backup FCV modulates to supply 400 gpm total.

[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.

When the 82% level is achieved, the primary FCV will maintain its previous position (200 gpm) and the backup FCV transfers to the Level Control Mode to maintain 82% level. As the level increases past the 82% wide range level, the backup FCV modulates until fully closed.

As steam generator level increases to 84% wide range, the primary FCV transfers to the Level Control Mode to maintain 85%. Level is maintained at 85% wide range level by the modulating primary FCV.

If an SIAS is actuated, the modulating control valves will be remain closed, if the level in the steam generator is above the "Critical Level". The modulating control valves will be switched immediately to the level control mode of operation, if the level in the steam generator is below the "Critical Level". The modulating control valves remain available for operator's manual control.

Operation with Backup Flow Control Valve (BFCV) (EFW-223 A or B) in Manual:
To resolve the condition, the BFCV will be placed in MANUAL, rather than AUTOMATIC as was done during this event. Placing the back-up valves in manual control effectively eliminated the flow feedback portion of the control loop. After EFW flow to the steam generators was steady and predictable. With the back-up controllers in manual, the system would perform as described in the system description except that the BFCV will not automatically respond, and remain in the closed position unless Operator action is taken, or when the SG level reaches 36.3%. At this point, the BFCV receive the priority override signal, and go to the full open position (with no SIAS present). The valve will close again when the signal is reset at 40%, unless operators change the valve position to maintain flow. With the controller in Manual, operators, may control the valve at any time, except when in PRIORITY OPEN.

New operator actions performed prior to a system being capable of performing the specified safety function 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 [Reference 4]. The manual operator action was evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 (addressed in Question #6) and found to meet the regulatory requirements. The operator action is allowed to be credited from the control room after 10 minutes from event initiation and each action requires 1 minute.

References

1. Waterford 3 Technical Specification Amendment 242.
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 IMC-326, OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS FOR CONDITIONS ADVERSE TO QUALITY OR SAFETY, 1/31/14.
5. ANSI/ANS-58.8-1994, American National Standard Time Response Design Criteria for Safety Related Operator Actions, August 23, 1994.
6. NEI 96-07 Revision 1, Guidelines for 10CFR50.59 Implementation, November 2000.
7. ANSI/ANS-58.9-1981, Single Failure Criteria for Light Water Reactors, February 17, 1981.

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 Steelman / per Telecom Paulette M. Robinson / SMI / Engineering / 6-6-15
Name (print) / Signature / Company / Department / Date

Reviewer: Michelle Groome / [signature] / Entergy / Engineering / 6-6-15
Name (print) / Signature / Company / Department / Date

OSRC: / Brian Lanka / [signature] 6-6-15
Chairman's Name (print) / Signature / Date

15-~~07~~ 06
OSRC Meeting #

R2K 6-30-15) BJL 6-30-15

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 does not affect any accident initiator. UFSAR Chapter 6 and 15 were reviewed to identify which accidents previously evaluated could be initiated or caused by the proposed change. UFSAR Chapter 15.1 (Increase in Heat Removal by the Secondary System) provides accidents in which the secondary side (Steam Generators) remove excess heat from the primary side (Reactor Coolant System). An inadvertent EFW actuation could add additional inventory to the Steam Generators (SGs). This change will have the OP-009-003 Steps 6.1.8 and 8.1.7 being deviated to allow Emergency Feedwater (EFW) Backup Flow Control Valves (EFW-223A(B)) controllers to be placed in manual. With EFW-223A(B) in manual, there is a lower frequency or possibility that an EFW feed event could occur, so there is not a more than minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The proposed change also does not create any new system interactions that could cause an accident. 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:

This change will have the OP-009-003 Steps 6.1.8 and 8.1.7 being deviated to allow Emergency Feedwater (EFW) Backup Flow Control Valves (EFW-223A(B)) controllers to be placed in manual. UFSAR Chapter 6 and 15 were reviewed to identify which accidents previously evaluated could increase the likelihood of a malfunction. The safety function of the EFW system is to provide sufficient supply of cooling water to one or both Steam Generators (SG) for the removal of decay heat from the Reactor Coolant System (RCS). For a malfunction, the analysis could be impacted by either less flow or too much flow.

UFSAR Chapter 15.1 (Increase in Heat Removal by the Secondary System) provides accidents in which the secondary side (Steam Generators) removes excess heat from the primary side (Reactor Coolant System). With EFW-223A(B) in manual, EFW-223A(B) will remain closed until either a PRIORITY OPEN MODE or manual operator action. The closed direction is the desired valve position to prevent or limit a UFSAR Chapter 15.1 increase in heat removal event. Therefore, these events do not have more than minimal increase in the likelihood of occurrence of a malfunction.

UFSAR Chapter 15.2 (Decrease in Heat Removal by the Secondary System) provides accidents in which the secondary side (Steam Generators) removes less heat from the

primary side (Reactor Coolant System). With EFW-223A(B) in manual, the EFW-223A(B) valves will still be driven open when the SG level falls to the "LO-LO LEVEL" of 36.3% Wide Range which is the PRIORITY OPEN MODE of operation (with no safety injection actuation signal present). The UFSAR Chapter 15 accident analyses do not credit EFW flow until the 36.3% WR level, so the EFW remains consistent with the analyses. The operator would be required to continue feeding the SGs once the priority open mode has been reset. The likelihood of a PRIORITY OPEN MODE malfunction is not impacted by this change.

The proposed change does not affect the likelihood of an equipment malfunction because the manual EFW control was already a manual operator action after 10 hours as described in UFSAR Section 10.4.9.

The proposed change also does not create any new system interactions that could cause an accident. 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:

This change will have the OP-009-003 Steps 6.1.8 and 8.1.7 being deviated to allow Emergency Feedwater (EFW) Backup Flow Control Valves (EFW-223A(B)) controllers to be placed in manual. UFSAR Chapter 6 and 15 were reviewed to identify which accidents previously evaluated could increase the accident consequences.

The safety function of the EFW system is to provide sufficient supply of cooling water to one or both Steam Generators (SG) for the removal of decay heat from the Reactor Coolant System (RCS). Emergency Feedwater (EFW) Backup Flow Control Valves EFW-223A(B) are fail open valves which means on a loss of motive air their safety function is in the open direction. EFW-223A(B) are in series with EFW Backup Flow Isolation Valves EFW-229A(B).

The UFSAR Chapter 15 accidents were evaluated to determine the impact of placing EFW-223A(B) in manual. The UFSAR Chapter 15 accident analyses do not credit EFW flow until the Steam Generator reaches 36.3% WR level. With EFW-223A(B) in manual, the EFW-223A(B) valves will still be driven open when the SG level falls to the "LO-LO LEVEL" of 36.3% Wide Range which is the PRIORITY OPEN MODE of operation (with no safety injection actuation signal present). The operator would be required to continue feeding the SG once the priority open mode has been reset. In addition, for a steam generator to not be fed with EFW-223A(B) in manual would require single failure of the parallel leg EFW path valve to close. For most Chapter 15 accident analyses, this failure would be less severe than those already analyzed.

Station blackout event will not be impacted by this action because for this event, only the Emergency Feedwater Pump AB is assumed available and only provides the designed required EFW flow.

For UFSAR Chapter 15 accidents which initiate a safety injection actuation signal (SIAS), with

EFW-223A(B) in manual, the PRIORITY OPEN MODE of operation will not work. Accidents which generate a SIAS are the UFSAR Chapter 15.1 and 15.6 type events. UFSAR Chapter 15.1 (Increase in Heat Removal by the Secondary System) provides accidents in which the secondary side (Steam Generators) removes excess heat from the primary side (Reactor Coolant System). With EFW-223A(B) in manual, EFW-223A(B) will remain closed until manual operator action. The closed direction is the desired valve position to prevent or limit a UFSAR Chapter 15.1 increase in heat removal event. Therefore, these events will not have the consequences increased. For the UFSAR Chapter 15.6, several events will be discussed. The Steam Generator Tube Rupture event already assumes manual action to initiate EFW due to this being more limiting consequence. UFSAR 15.6.3.2.1.2 states:

> An automatic Emergency Feedwater Actuation Signal (EFAS) is generated if the level in the SG falls below 27.4% NR (71% WR). However, Emergency Feedwater (EFW) flow to the SG will not commence until the level falls below the Critical Level (55% WR). Since the level had not fallen below the Critical Level at seven minutes after reactor trip, EFW flow had not yet been initiated. Thus, it was assumed that the operator would take manual control of the EFW system and initiate EFW flow subsequent to that time.

For the steam generator tube rupture, the early EFW initiation and associated cooldown produces more adverse consequences. So for any operator action that would occur later in time, would produce less severe consequences.

Large break Loss of Coolant Accident (LOCA) does not require steam generators for heat removal so that event is not impacted. The Small Break LOCA does require secondary heat removal. The Small Break LOCA calculation (CN-OA-08-61) does not address EFW flow but does address atmospheric dump valve and main steam safety valve operation. Small Break LOCA calculation (CN-OA-08-61) Appendix C shows the analysis figures. The steam generator steam flow shows that a small amount of inventory is used out of each steam generator. The operator action to manual control EFW flow via EFW-223A(B) would produce accident results consistent with that of the current design analysis. Thus, the UFSAR Chapter 15.6 events are not adversely impacted by this change.

For the potential to overfill the steam generator due to operator action, calculation ECS02-005 (Effects of Uncontrolled EFW on Accident Situations) already credits operator action to either stop EFW flow or close the impacted steam generator isolation valve when the high steam generator level alarm is actuated. This change will not impact this event.

Technical Specification (TS) 3.3.2 requires the Emergency Feedwater Actuation Signal to be Operable, including Automatic Actuation Logic and Control Valve Logic (Wide Range SG Level Low). The Automatic Actuation Logic of EFAS operated properly following the plant trip (EFAS automatically actuated as designed). The specific instrumentation and components that constitute the TS 3.3.2 Control Valve Logic have been previously defined in EC-31402. EC-58103 confirmed that the valve oscillations which occurred did not affect the safety function of the Control Valve Logic components as defined for TS 3.3.2.

TS 3.7.1.2 requires three emergency feedwater pumps and two flow paths to be operable in Modes 1, 2, and 3. The equipment required in TS 3.7.1.2 remain capable of performing their safety function with the EFW-223A(B) in manual and crediting operator action.

The TS surveillance requirements continue to be met.

The proposed change will ensure EFW will be available in a timely manner and capable of performing its specified function. 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:

This change will have the OP-009-003 Steps 6.1.8 and 8.1.7 being deviated to allow Emergency Feedwater (EFW) Backup Flow Control Valves (EFW-223A(B)) controllers to be placed in manual. UFSAR Chapter 6 and 15 were reviewed to identify which accidents previously evaluated could increase the accident consequences.

UFSAR Chapter 15.1 (Increase in Heat Removal by the Secondary System) provides accidents in which the secondary side (Steam Generators) removes excess heat from the primary side (Reactor Coolant System). With EFW-223A(B) in manual, EFW-223A(B) will remain closed until either a PRIORITY OPEN MODE or manual operator action. The closed direction is the desired valve position to prevent or limit a UFSAR Chapter 15.1 increase in heat removal event. Therefore, these events not have the consequences increased.

UFSAR Chapter 15.2 (Decrease in Heat Removal by the Secondary System) provides accidents in which the secondary side (Steam Generators) removes less heat from the primary side (Reactor Coolant System). With EFW-223A(B) in manual, the EFW-223A(B) valves will still be driven open when the SG level falls to the "LO-LO LEVEL" of 36.3% Wide Range which is the PRIORITY OPEN MODE of operation (with no safety injection actuation signal present). The UFSAR Chapter 15 accident analyses do not credit EFW flow until the 36.3% level, so the EFW remains consistent with the analyses. The operator would be required to continue feeding the SGs once the priority open mode has been reset. The credit of operator action (refer to question 6 for the regulatory requirements) ensure EFW is available and a secondary heat sink remains. Thus, this change does not result in more than a minimal increase in the consequences of a malfunction.

The proposed change also does not create any new system interactions that could cause an accident. Therefore, the proposed change does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

☐ Yes
☒ No

BASIS:

This change will have the OP-009-003 Steps 6.1.8 and 8.1.7 being deviated to allow Emergency Feedwater (EFW) Backup Flow Control Valves (EFW-223A(B)) controllers to be placed in manual. UFSAR Chapter 6 and 15 were reviewed to identify which accidents previously evaluated could create a different accident.

The EFW system is primarily an accident mitigation system. The function and use of the EFW system will remain the same. Thus, an accident of a different type than previously evaluated 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:
This proposed change does not impact the EFW configuration or the EFW system components. 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.

If an operator action must be performed prior to a system being capable of performing it 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 [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 an Unreviewed Safety Question (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 is evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994. The ten primary attribute evaluations are specifically listed below.

(1) The specific operator actions required (procedures);

The specific operator action will be to control EFW-223A(B) in manual during a design basis event. Item (4) below gives the specific procedures that will instruct this action. No adverse impact.

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

The operation of EFW-223A(B) will be from the control room or the remote shutdown panel (LCP43). LCP43 is only credited for a control room fire type event and is included for completeness. Thus, potentially harsh or inhospitable environmental remain within the current design basis. No adverse impact.

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

The operation of EFW-223A(B) will be from the control room or LCP43. Thus, the ingress/egress paths are those within the current design basis. No adverse impact.

(4) The procedural guidance for required actions;

The operator actions required to control EFW during a design basis accidents are contained in the Emergency Operating Procedures. Immediate operator actions are contained in OP-902-000 and then optimal or functional emergency operating procedures are entered via procedures OP-902-001 through OP-902-008.

OP-902-000 (Standard Post Trip Actions) provides the immediate operator actions. Step 6 provides a required Steam Generator level band and verification the EFW is available to restore inventory into the required band.

OP-902-001 (Reactor Trip Recovery Procedure) is for an uncomplicated reactor trip with feedwater available. EFW is not needed in this optimal procedure. Step 9 provides the required Steam Generator level band.

OP-902-002 (Loss of Coolant Accident Recovery) Step 31 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-003 (Loss of Offsite Power / Loss of Forced Circulation Recovery) Step 10 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-004 (Excess Steam Demand Recovery) first isolates the impacted Steam Generator then Step 16 uses EFW in manual to feed the unaffected Steam Generator. Step 22 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-005 (Station Blackout Recovery) Step 10 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-006 (Loss of Main Feedwater Recovery) Step 11 gives options for the feedwater recovery. Step 13 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-007 (Steam Generator Tube Rupture Recovery) Step 16 and 17 determines and isolates the most affected Steam Generator. Step 26 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-008 (Functional Recovery Procedure) provides multiple success path which include a Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

The emergency operating procedures already contain the needed information to instruct the operator on the required actions. The emergency operating procedures would bound those actions prescribed in the off-normal procedures with the exception of OP-901-502 (Evacuation of Control Room and Subsequent Plant Shutdown). OP-901-502 Attachment 3 (At the control time critical action) Step 4 already places EFW-223A(B) in manual to control steam generator level within a procedurally prescribed band. No adverse impact.

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

The operator action will be performed from the control room or LCP43 in accordance with the operations procedures listed in Item (4). The operator will be required to be licensed and current on their license requirements. The operating procedures are those trained on during requalification cycles so no additional training is required. A licensed operator is already the minimum requirement to perform control board actions. In addition, an operations standing instruction will require each operator taking the dedicated assignment to be brief prior to standing watch. No adverse impact.

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

In order to ensure the action is completed during a design basis accident, an additional dedicated operator will be utilized for this action. During a design basis accident, multiple distractions can occur with only the Technical Specification minimum staffing available. To ensure this action is properly handled an additional dedicated operator will be assigned to this task. The dedicated operator only emergency operating task will be to perform the EFW manual action. With being dedicated to only this task, the procedure actions will be accomplished much quicker than the analysis credited times. 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 determination of whether operator action is required begins with the entry into the OP-902-000 standard post trip actions. The operator will use indications on control room panel CP-8 (or LCP43) which are control room design accident indications for steam generator level and emergency feedwater response. The steam generator levels restoring to the procedural bands will be indication of the current response. These parameters are also monitored by other members of the control room staff (control room supervisor, shift technical advisory, balance of plant operator). 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;

ANSI/ANS-58.9-1981 [Reference 7] describes the requirements for single failures during design basis accidents. ANSI/ANS-58.9-1981 Section 3 rules states the a design basis accident plus single active failure is required to be analyzed. ANSI/ANS-58.9-1981 Section 3.7 states that an operator error is considered a single active failure.

ANSI/ANS-58.8-1994 Section 3.1.3 states safety-related operator actions shall be credited only where a single operator error of omission does not result in exceeding any limiting design requirement for the DBE under consideration.

For design basis accidents, the EFW system remains capable of performing its specified function with EFW-223A(B) being manually controlled. The failure of the operator to perform the credited operator function on a single backup flow control valve would be no different than currently analyzed single active failures required by ANSI/ANS-58.9-1981. To be more specific, the operator error could be an omission or a wrong action, and the operator error would still be considered an active single failure per ANSI-58.9-1981. In general, Design Basis Accidents have an event initiator and assume one single active failure. If the operator action is the single failure, then no additional single failure would have to be assumed. The Design Basis Accidents have identified their limiting single failures with respect to a specific acceptance criteria, the potential operator action failure would be no more adverse than that already analyzed in the FSAR.

During EFAS with a SIAS, if the steam generator level reaches 36.3%, the valves will not

receive the PRIORITY OPEN signal. The SIAS will block the PRIORITY OPEN signal to all flow control valves, and the valves will remain in Auto or Manual (according the position on the control station), and remain the current valve position (modulating if in AUTO, or Fixed if in Manual). The SIAS will only block the PRIORTY OPEN signal, and the operator can adjust EFW flow as necessary during the event with the SIAS present. This SIAS override of the PRIORITY OPEN signal is the current design, which allows the operator to control EFW flow under SIAS conditions, and is unchanged by this Manual Operator Action.

IMC-326 [Reference 4] provides the following guidance:
Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. Accordingly, it is not appropriate to consider SSCs operable by taking credit for manual action in place of automatic action for protection of safety limits.
Technical Specification Section 2 provides the limiting safety system settings and safety limits. The EFW flow control valve setpoints are not TS Section 2 limiting safety system settings or safety limits, thus this change is not excluded upon this criteria.

The ability to recover from creditable errors are covered by automatic system actions and control room alarms. If the operator does nothing and does not provide inventory to the steam generators. EFW-223A(B) valves will still be driven open when the SG level falls to the "LO-LO LEVEL" of 36.3% Wide Range which is the PRIORITY OPEN MODE of operation (with no safety injection actuation signal present). If the operator provides too much flow, the steam generator high alarm will annunciate alerting all the control room operators of the need to secure EFW flow.

As defense in depth, a caution tag is being placed on each control room switch to signify the action. In addition to the dedicated operator, the control room supervisor, shift technical advisory, balance of plant operator will also be validating plant conditions and are able to identify and correct any credible error. The balance of plant operator has historically completed OP-902-000 standard post trip actions within the 11 minutes that an operated is credited to perform manual actions.

No adverse impact.

(9) Consideration of the risk significance of the proposed operator actions;
Considering the availability of cues to the operators, the non-adverse environment, locality of the action, available procedures, and qualified personnel, the impact of this change on plant risk is expected to be negligible. The risk significance for EFW initiation has not changed. The EFW will still actuate when required and the operator action ensures sufficient feeding to maintain steam generator desired levels. 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 provide time requirements for different accident scenarios. For a EFW malfunction using the ANSI/ANS-58.8-1994 guidance 10 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 which is not required for this condition. A dedicated operator will be stationed in the control room envelope to perform the required procedural actions. 10 minutes will preserve the required EFW motive force to ensure the safety function continues to be met.

The accident analyses assumes a flow of 575 gpm to the generator when the level reaches 36.3%, when the flow control valves go to the full open position. Based upon CE-002-001, there is 1047 gallons of water needed to fill a SG from 36.3 to 40%. Calculation ECM88-024 documents 20 complete strokes of the EFW back up flow control valve over 10 hours for the accumulator sizing. Using this information, there is approximately 36 minutes before the operator must take manual control of back up EFW flow control valve. At this point in time, the Nitrogen Accumulator credited volume for the backup flow control valve would be used and thus begin limiting other valves strokes. Therefore operator action within 11 minutes is adequate to preserve the nitrogen accumulator volume.

Since, the regulatory requirements have been met to credit the operator action. This 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:

This change will have the OP-009-003 Steps 6.1.8 and 8.1.7 being deviated to allow Emergency Feedwater (EFW) Backup Flow Control Valves (EFW-223A(B)) controllers to be placed in manual. This could impact the amount of water and timing of inventory to the steam generators but does not specifically impact the fuel clad, reactor coolant system boundary, or containment. This ensures that the current limiting safety analysis for dose consequences and fission product barrier limits remain bounding. 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.

For the potential to overfill the steam generator due to operator action which could cause a failure of the main steam line piping, calculation ECS02-005 (Effects of Uncontrolled EFW on Accident Situations) already credits operator action to either stop EFW flow or close the impacted steam generator isolation valves when the high steam generator level alarm is actuated. This change will not impact this event.

Steam generator level above the tube sheet is credit for radiological scrubbing for the dose analysis. With credit of the operator action, the steam generator water level and feed rates will be consistent with the current design basis. Thus, the radiological consequences will not be impacted.

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

BASIS:

Methods of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC [Reference 6 Section 3.10]. This change is not impacting the methodology or topical reports used in the UFSAR analyses. The existing UFSAR evaluations utilizing EFW as a mitigating system have 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.

I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # / Rev. #:** 15-01 / R1

Proposed Change / Document: CR-WF3-2015-3565 Operability/Compensatory Measure.

Description of Change: The Emergency Feedwater (EFW) backup flow control valves (EFW-223A(B)) have been changed from automatic to manual operation. With EFW-223A(B) in manual control, compensatory control room manual operator action is required to ensure these valves can be credited to meet their safety function. This evaluation demonstrates that a dedicated operator can control EFW-223A(B) from the control room within 30 minutes. This revision improves the documentation and changes the operator required action time to within 30 minutes.

Summary of Evaluation:
This 50.59 evaluation addresses crediting a compensatory control room manual operator action to control Emergency Feedwater (EFW) backup flow control valves (EFW-223A(B)). Waterford 3 acknowledges that the control room manual action is expected to be a temporary measure and to promptly end when the automatic action is corrected in accordance with 10CFR50 Appendix B and the corrective action program [NRC Inspection Manual IMC-326 Reference 4]. This condition has been entered into the corrective action program under CR-WF3-2015-3565.

CR-WF3-2015-3565 documented that EFW-223A(B) cycled more than assumed in the nitrogen accumulator sizing calculation [ECM88-024 Reference 16]. The additional valve strokes could exhaust the safety related nitrogen accumulators prior to the credited 10 hours of remote valve operation. With EFW-223A(B) in manual control, the automatic valve oscillations experienced in CR-WF3-2015-3565 were eliminated.

The EFW system at Waterford 3 consists of two (50 percent capacity) motor-driven pumps, and one (100 percent capacity) turbine-driven pump. Steam is provided to the turbine-driven EFW (TDEFW) pump from each of the two steam generators (SGs) through separate flow paths. All three pumps are connected to a common discharge header. The EFW flow path for each SG is designed to transport EFW flow from the common EFW pump discharge header to the main feedwater line (downstream of the main feedwater isolation valves) through two parallel legs. Each parallel leg contains a flow control valve and an isolation valve in series. These valves are fail-open pneumatic valves. Safety-related nitrogen accumulators serve as backup to the instrument air system for these valves. Each nitrogen accumulator supplies a pair of EFW valves (one flow control valve and one isolation valve) in different parallel legs to the same SG. One flow path through either of the two parallel legs to a SG is capable of providing 100 percent of the minimum required flow to its associated SG and performing the required decay heat removal safety function without reliance on the other SG.

The EFW system Safety Function is:

1. To provide sufficient supply of cooling water to the steam generators (SG) for the removal of decay heat from the reactor during emergency situations when the main feedwater system is not available, and
2. To maintain the SG level above the top of the SG tubes for mitigation of radiological dose consequences to the control room operators and general public.

The operator is performing the safety related function of controlling EFW flow via EFW-223A(B). The failure of the operator to perform this function either by omission or wrong action is considered an

[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.

active single failure [ANSI/ANS-58.9-1981 Reference 7]. In general, design basis accidents have an event initiator and assume one active single failure. Each steam generator has two EFW flow paths available to provide sufficient inventory to the steam generators. The design basis accidents have identified their limiting single failures with respect to a specific acceptance criteria, the potential operator action failure would be no different than the active single failure of one or both of the EFW backup flow control valves. With the failure of one or both EFW backup flow control valves, the EFW primary flow control valves would still be available to perform the EFW safety function. Therefore, the operator action failure is no more adverse than that already analyzed in the UFSAR.

Furthermore, the EFW system is designed for, and operating procedures specify, either manual or automatic control as the means to perform the EFW safety function during an accident. The safety analyses already credit manual operator control of EFW flow, typically after 30 minutes. Therefore, manual operator control of the EFW backup flow control valves does not introduce any new feature, function, or control method to plant equipment.

New operator actions performed prior to a system being capable of performing the specified safety function 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 [Reference 4]. The control room manual operator action was evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 [Reference 5] (addressed in Question #6) and found to meet the regulatory requirements. The operator action is allowed to be credited from the control room.

ANSI/ANS-58.8-1994 was used to determine the control room operator action time of 30 minutes was allowable.

<u>References</u>

1. Waterford 3 Technical Specification through Amendment 242.
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 IMC-326, OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS FOR CONDITIONS ADVERSE TO QUALITY OR SAFETY, 1/31/14.
5. ANSI/ANS-58.8-1994, American National Standard Time Response Design Criteria for Safety Related Operator Actions, August 23, 1994.
6. NEI 96-07 Revision 1, Guidelines for 10CFR50.59 Implementation, November 2000.
7. ANSI/ANS-58.9-1981, Single Failure Criteria for Light Water Reactors, February 17, 1981.
8. CN-TAS-08-40 R0, Waterford 3 Feedwater Line Break Analysis for RSGs, June 4, 2009.
9. CN-SEE-II-09-21 R3, Natural Circulation Cooldown to 350F to Support BTP 5-4 Criteria for Waterford 3 with Replacement Steam Generators, September 15, 2014.
10. EC58184, EFW Response Time Justification, June 12, 2015.
11. NUREG-0787, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, July 1981.
12. CPENG96-121, Design Basis Reconstitution for Emergency Feedwater Flow Rate: Investigation of Steam Line Break, June 27, 1996.
13. CN-TAS-08-30 Revision 1, Waterford 3 Post Trip Main Steam Line Break Analysis for RSGs, March 26, 2010.
14. CN-TAS-03-30 Revision 5, Waterford 3 Chapter 15 EAB and LPZ Dose Consequences.
15. MNQ10-1 Revision 3, Emergency Feedwater System Head Curves.

16. ECM88-024 Revision 3, Nitrogen Accumulator V, VIII, IX and X Sizing Calculation.
17. ECS02-005 EC8458, Effects of Uncontrolled Maximum EFW on Accident Situations, April 5, 2010.
18. Regulatory Guide 1.53 Revision 2, Application of the Single Failure Criterion to Safety Systems, November 2003.
19. IEEE 379-2000, IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems, September 21, 2000.
20. NUREG-0800 APPENDIX 18-A Revision 0, CREDITING MANUAL OPERATOR ACTIONS IN DIVERSITY AND DEFENSE-IN-DEPTH ANALYSES, April 2014 [NRC ADAMS Accession Number ML13115A156].
21. OP-902-000 Revision 15, Standard Post Trip Actions, December 30, 2012.
22. OP-902-001 Revision 14, Reactor Trip Recovery Procedure, December 30, 2012.
23. OP-902-002 Revision 19, Loss of Coolant Accident Recovery, February 24, 2015.
24. OP-902-003 Revision 9, Loss of Offsite Power / Loss of Forced Circulation Recover, February 24, 2015.
25. OP-902-004 Revision 15, Excess Steam Demand Recovery, February 24, 2015.
26. OP-902-005 Revision 18, Station Blackout Recovery Procedure, February 24, 2015.
27. OP-902-006 Revision 16, Loss of Main Feedwater Recovery, February 24, 2015.
28. OP-902-007 Revision 16, Steam Generator Tube Rupture Recovery, February 24, 2015.
29. OP-902-008 Revision 23, Functional Recovery Procedure, February 24, 2015.
30. OP-901-502 Revision 30, Evacuation of Control Room and Subsequent Plant Shutdown, April 28, 2015.
31. OP-901-502-01 Revision 2, Time Critical Task Resource Management for Control Room Evacuation, February 12, 2015.
32. OI-042-000 Revision 37, Watch Station Processes, July 23, 2015.

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 Steelman / [signature] Samad per telecon clav 16 / SMI / Engineering / 7-31-15
Name (print) / Signature / Company / Department / Date

Reviewer: Dale Gallodoro / [signature] / SMI / Engineering / 7-31-15
Name (print) / Signature / Company / Department / Date

OSRC: / Brian Lanka / [signature] / 7-31-15
Chairman's Name (print) / Signature / Date

15-13
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:

This change addresses crediting a compensatory control room manual operator action to control Emergency Feedwater (EFW) backup flow control valves (EFW-223A(B)). UFSAR Chapter 6 and 15 were reviewed to identify which accidents previously evaluated could be initiated or caused by the proposed change. The applicable UFSAR section is UFSAR Chapter 15.1 (Increase in Heat Removal by the Secondary System) which provides accidents in which the secondary side (Steam Generators) remove excess heat from the primary side (Reactor Coolant System). An inadvertent EFW actuation could add additional inventory to the Steam Generators (SGs). UFSAR Section 15.1.1.2 addresses this accident caused by the startup of emergency feedwater system. UFSAR Section 15.1.1.2 assumes the EFW is actuated and cools down the secondary system.

With EFW-223A(B) in manual, the frequency of a spurious EFW operation has not changed because no physical change to the plant has occurred. The EFW controls are still designed to the same standards and requirements as previously. The frequency of an operator error to initiate an accident has not changed because EFW is not utilized for normal operation and without an Emergency Feedwater Actuation Signal (EFAS), the operation of the EFW backup flow control valves would not add inventory to the steam generators due to no EFW pumps running and the EFW flow isolation valves being closed. The proposed change does not affect any of the accident initiators.

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:

This change addresses crediting a compensatory control room manual operator action to control Emergency Feedwater (EFW) backup flow control valves (EFW-223A(B)). The failure of the operator to perform this function either by omission or wrong action is considered an active single failure [ANSI/ANS-58.9-1981 Reference 7]. In general, design basis accidents have an event initiator and assume one active single failure. Each steam generator has two EFW flow paths available to provide sufficient inventory to the steam generators. The design basis accidents have identified their limiting single failures with respect to a specific acceptance criteria, the potential operator action failure would be no different than the active single failure of one or both of the EFW backup flow control valves. With the failure of one or both EFW backup flow control valves, the EFW primary flow control valves would still be available to perform the EFW safety function.

The operator has sufficient time, information (control board), and procedures to perform this action. The following items describe why it is unlikely that the operator would not adequately perform this action.

- Caution tags have been added to the EFW backup flow control valves to aid in identifying the correct valves. It is unlikely with this aid that the operator would not recognize the valves to operate.
- CR-WF3-2011-1714 CA#30 was used to determine the time for all crews to complete OP-902-000 (Standard Post Trip Actions). All crews (100%) consistently completed OP-902-000 actions in less than 10 minutes during a Station Blackout timed evaluation. EC58184 validated that at 30 minutes, the EFW backup flow control valves will still have 11 out of 20 strokes available to perform their safety function. It is unlikely that with this time available, the operator would fail to take the correct action.
- The operator will use indications on control room panel CP-8 (or LCP43) which are control room design accident indications for steam generator level and emergency feedwater response. The Emergency Operating Procedures (EOPs) contain the needed information to instruct the operator on the required actions to maintain steam generator required levels. It is unlikely that with the steam generator indications that the operator would fail to follow the procedure guidance to maintain steam generator levels within the required band.
- The control room has multiple staff all validating the plant safety functions. It is unlikely that the control room supervisor, at the controls operator, balance of plant operator, shift technical advisor, and shift manager would all fail to follow procedures and validate the Reactor Coolant System Heat Removal safety function.
- Control room annunciators will alarm at lowering steam generator levels. It is unlikely that the control room staff would not utilize these indications to take corrective actions.
- EC58184 calculated that 69.9 minutes is available prior to all 20 EFW backup control valve strokes being used using the design basis volumes. This means margin exists between that credited in this evaluation and the time when the nitrogen accumulator volume reserved for the EFW backup control valves will be consumed. It is unlikely with this extra available time that the dedicated operator and operations crew would all fail to follow their procedures and maintain steam generator levels.
- As defense in depth information, the nitrogen accumulator sizing calculation (ECM88-024) shows that only 1% of the stored nitrogen is dedicated for stroking EFW-223A(B) and more than 50% is reserved for potential leakage. The most recent surveillance test demonstrated that actual leakage is less than half the allowed leakage. With this available nitrogen accumulator margin, it would take over 6 hours prior to the accumulator depletion. It is unlikely with this extra available time that the dedicated operator and operations crew would all fail to follow their procedures and maintain steam generator levels.

The operator has sufficient time, information (control board), and procedures to perform the required action, thus this change does not result in more than a minimal increase in the likelihood of the operator failing to control EFW flow.

This change does not change any of the physical structures, systems, or components. The EFW system was designed to allow manual control of the EFW backup flow control valves. Some safety analyses currently assume operator control of EFW flow, typically after 30 minutes. UFSAR Section 10.4.9 already identified that after 10 hours manual

EFW control was a required manual local operator action. The proposed change also does not create any new physical system interactions that could cause a malfunction. Thus, the proposed change does not result in more than a minimal 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:

This change addresses crediting a compensatory control room manual operator action to control Emergency Feedwater (EFW) backup flow control valves (EFW-223A(B)). UFSAR Section 7.3.1.1.6.1 describes that the EFS instrumentation and controls are designed for automatic operation during emergency situations such as steam pipe rupture, loss of normal feed, and plant black out. The emergency feedwater control valves may be operated by operator manually or may be left in automatic mode of control. UFSAR Section 7.3.1.1.6.2 gives a description of the automatic EFW Control system operation on regulating the flow of EFW to the steam generators so as to minimize adverse effects on the Reactor Coolant System. The automatic EFW control system is credited in applicable UFSAR Chapter 15 analyses and is discussed in UFSAR Section 10.4.9.3.6.

Figure Q3-1. Emergency Feedwater System

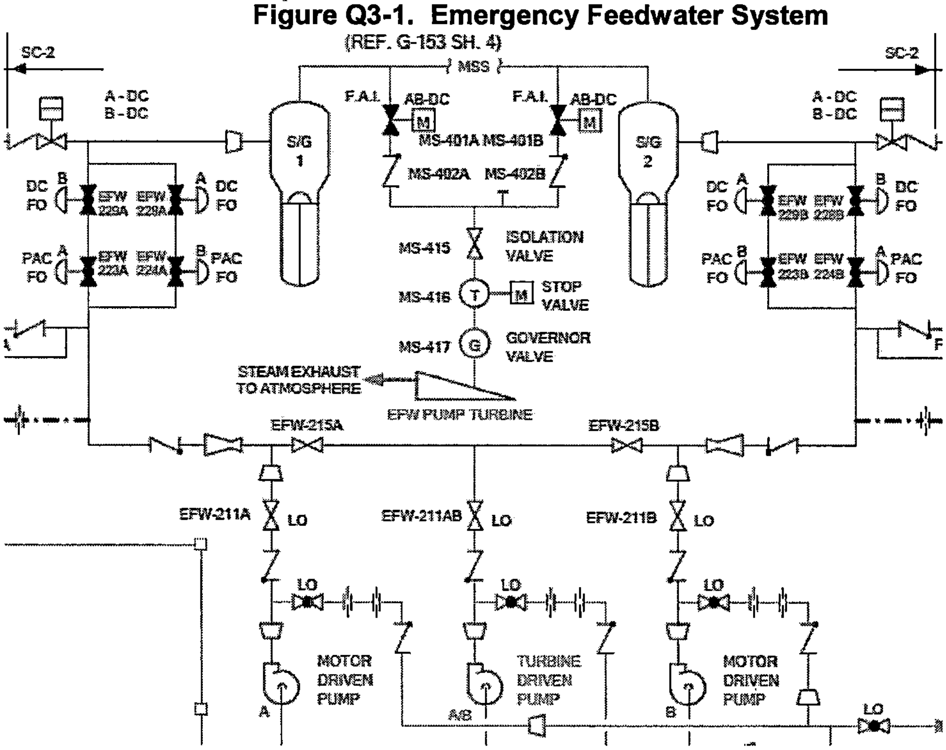

Figure Q3-1 shows that each steam generator has two (2) parallel EFW flow paths with each flow path capable of providing 100% of the required EFW flow. For design basis events which have two (2) steam generators available for heat removal, the flow paths through the primary flow control valves (EFW-224A(B)) would be sufficient to ensure a secondary heat sink. If a primary flow control valve (EFW-224A or B) is considered as

the limiting single failure, then conservatively no flow to that steam generator would occur until operator action to open EFW-223 A or B but the other steam generator would have full flow from 3 EFW pumps (~200% capacity) [Reference 15 Section 2]. For design basis events with two (2) steam generators available as a heat sink, the control room operator action to control the EFW backup flow control valves (EFW-223A(B)) would not be time critical as the EFW primary flow control valves (EFW-224A(B)) provide sufficient redundancy to ensure a secondary heat sink remains available.

For design basis events which have one (1) steam generator available for heat removal, the event consequences need to be validated to ensure they remain within the acceptance limits. The following design basis events are initiated with an accident that causes a malfunction with one steam generator, these are assessed for potential impacts:

UFSAR 15.1.1.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

This event is caused by an atmospheric dump valve that may have been inadvertently opened by an operator or may have been opened due to failure in the control system. This event does not assume an active single failure, thus the EFW primary flow control valve (EFW-224A(B)) on the unaffected steam generator would be available to provide full EFW flow.

UFSAR Table 15.1-3 states that at 1800 seconds (30 minutes) the operator takes control of the plant.

Thus, this event is not impacted by this change.

UFSAR 15.1.2.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component

The inadvertent opening of a steam generator atmospheric dump valve with loss of offsite power is a cooldown event modeled to ensure sufficient thermal margin is preserved to prevent a Specified Acceptable Fuel Design Limit (SAFD) violation. UFSAR Section 15.1.2.4.3.3 states that should an Emergency Feedwater Actuation Signal (EFAS) occur, EFW would not adversely affect DNB performance or releases to the environment. Consequently, EFW was not modeled prior to 1800 seconds.

UFSAR Table 15.1-8C states that at 1800 seconds (30 minutes) the operator takes control of the plant.

Thus, this event is not impacted by this change.

UFSAR 15.1.3.1 Steam System Piping Failures Post-Trip Return-To-Power
UFSAR 15.1.3.2 Steam System Piping Failures Inside and Outside Containment Modes 3 and 4 with All CEAs on the Bottom
UFSAR 15.1.3.3 Steam System Piping Failures: Pre-Trip Power Excursion Analysis

The steam line break events are assessed together due to similar event characteristics. The steam line break events are analyzed as cooldown events to minimize DNBR and maximize Peak Linear Heat Rate (PLHR) to determine any potential fuel failure. Since these are cooldown events, the earlier the EFW flow occurs the more adverse to the transient. Thus, an EFW flow delay due to the operator action timing would be beneficial for this analyzed event. The long term system response would be bounded by the UFSAR Section 15.2.3.1 (Feedwater System Pipe Breaks) in terms of EFW requirements. In terms of radiological consequences, the delay in EFW flow to the

unaffected steam generator would have negligible impact because the dose consequences are dominated by the affected steam generator.

For a situation where the operator initiated EFW flow early in the event, the results would not be adversely impacted because the cooldown is dominated by the affected steam generator. The steam line break events cause a main steam isolation signal (MSIS) which would prevent the operator from initiating EFW flow to the affected steam generator. The original steam line break analysis submitted and approved by the NRC investigated multiple single failures. NUREG-737 Section 15.3.1 (Steam Line Breaks) [Reference 11] gives the following information:
A study of single failures was performed to determine which is most limiting. Failures considered included failure of main feedwater isolation valve to close after a main steam isolation signal (MSIS), failure of one main steam isolation valve to close after MSIS, failure of turbine stop valves to close after reactor trip, failure of one diesel generator to start after loss of offsite ac power, and failure of one high pressure safety injection (HPSI) pump to start after a safety injection actuation signal (SIAS). This study showed that loss of one HPSI pump had the most adverse effect.

The failure of the main feedwater isolation valve to close on the MSIS would be similar to the overfeeding with EFW. The HPSI pump failure is more adverse because the negative reactivity contribution from the injected boron is more important than the minor cooldown caused by the extra flow.

The impact of maximum EFW flow at the beginning of the steam line break event was investigated in CPENG96-121 [Reference 12]. CPENG96-121 contained attachment WS-FE-0147 which analyzed the impact of 2300 gpm EFW flow on the steam line break transient. The hot full power conclusion was the increased EFW flow had a negligible effect on the hot full power (HFP) post-trip steam line break transient scenarios during return to power because the event severity is dependent on steam generator dryout time and not EFW flow. The hot zero power (HZP) steam line break conclusion is that the initial margin is significantly larger than the hot full power steam line break, and the increased cooldown does not adversely impact that margin.

The current steam line break Analysis of Record (AOR) is CN-TAS-08-30 [Reference 13]. The AOR comment question #87 (page 357) discusses the impact of EFW flow on the steam line break transient. The conclusion is that the maximum EFW has a negligible impact on the event results.

UFSAR Table 15.1-12, 15.1-13, 15.1-14A, 15.1-14B, and 15.1-2 states that at 1800 seconds (30 minutes) the operator initiates cooldown.

Thus, this event is not impacted by this change.

UFSAR 15.2.3.1 Feedwater System Pipe Breaks

Calculation CN-TAS-08-40 R0 (Waterford 3 Feedwater Line Break Analysis for RSGs) is the analysis of record (AOR) for the feedwater line break events [Reference 8]. The AOR analyzes small and large feedwater line breaks with short and long term responses. The short term responses are focused on primary and secondary peak pressures. The long term response ensures no pressurizer fill and liquid through the primary safety valve. The large feedwater line break short term transient sequence of events is shown in Table 5.2.1-3. Table 5.2.1-3 shows no EFW flow is initiated prior to the maximum primary and secondary pressures being achieved. The small feedwater line break short

term transient sequence of events is shown in Table 5.2.2-3. Table 5.2.2-3 shows no EFW flow is initiated prior to the maximum primary and secondary pressures being achieved. The large feedwater line break long term transient sequence of events is shown in Table 5.2.3-2. Table 5.2.3-2 shows no EFW flow is credited in the event which ends at 1800 seconds (30 minutes). Attachment C Figures C.III-14 and C.III-15 show the steam generator levels and EFW flow (no flow). The small feedwater line break long term transient sequence of events is shown in Table 5.2.4-3. Table 5.2.4-3 shows no EFW flow is credited in the event which ends at 1800 seconds (30 minutes). Attachment C Figures C.IV-14 and C.IV-15 show the steam generator levels and EFW flow (no flow). The feedwater line break events do not require EFW flow until after 30 minutes.

UFSAR Table 15.2-8 and 15.2-9B states that at 1800 seconds (30 minutes) the operator takes control of the plant.

Thus, this event is not impacted by this change.

UFSAR 15.6.3.2 Steam Generator Tube Rupture

UFSAR Section 15.6.3.2.1.1 states:

The analysis of the radiological consequences of the SGTR to account for the impact of potential uncovery of the rupture during the event. Since greater radiological releases result if early operator actions are assumed, the first operator action was postulated to occur at 7 minutes after reactor trip. This is consistent with ANSI/ANS-58.8-1984, 'American National Standard Response Design Criteria for Nuclear Safety Related Operator Actions.'

UFSAR Section 15.6.3.2.1.2(1) states:

Since the level had not fallen below the Critical Level at seven minutes after reactor trip, EFW flow had not yet been initiated. Thus, it was assumed that the operator would take manual control of the EFW system and initiate EFW flow subsequent to that time.

This event already conservatively assumed the manual action to initiate EFW flow (UFSAR Table 15.6-24), thus there is no impact from this change.

UFSAR Chapter 15.1 (Increase in Heat Removal by the Secondary System) provides accidents in which the secondary side (Steam Generators) remove excess heat from the primary side (Reactor Coolant System). An inadvertent EFW actuation could add additional inventory to the Steam Generators (SGs). UFSAR Section 15.1.1.2 addresses this accident caused by the startup of emergency feedwater system. This change would have no impact on this event because it already assumes the EFW is actuated and cools down the secondary system.

The final impact needed to be addressed is the potential delay in initiating a reactor cooldown on EFW inventory. UFSAR Section 9.3.6.3.3 requires a natural circulation cooldown analysis to meet the requirements of NRC Branch Technical Position (BTP) 5-4. The BTP 5-4 AOR is CN-SEE-II-09-21 [Reference 9]. BTP 5-4 requires the analysis to assume a hold period of four (4) hours prior to commencing cooldown. The allowable operator action time credit is well within the 4 hour hold time, thus no impact from this change.

The failure of the operator to take any action to control EFW backup flow control valves (EFW-223A(B)) would be the event single active failure (common mode). The EFW primary flow control valves (EFW-224A(B)) would then be available to perform the EFW safety function to provide flow. The UFSAR and this evaluation still credits the operator at 30 minutes to use the available valves (EFW primary or backup flow control valves)

to maintain steam generator level and to cool the plant to shutdown cooling entry conditions. This is consistent with the Waterford 3 licensing basis. With the available EFW primary flow control valves maintaining level, the EFW backup flow control valves would not cycle at the "lo-lo level" and the safety related nitrogen accumulator volume would remain available.

This EFW manual action is similar to the shutdown cooling initiation. Initiating shutdown cooling is a manual operator action and no common mode failure is assumed because multiple staff members are available to perform this action, control room indications are available to demonstrate whether the action was taken, based upon the control room indications any failures can be corrected, and the actions are procedurally driven.

Regulatory Guide 1.53 [Reference 18] provides methods acceptable to the NRC with respect to single failure criterion. Regulatory Guide 1.53 endorses IEEE 379-2000 [Reference 19]. IEEE 379-2000 Section 5.5 addresses common cause failures. IEEE 379-2000 specifically states:

> Common-cause failures not subject to single-failure analysis include those that can result from external environmental effects (e.g., voltage, frequency, radiation, temperature, humidity, pressure, vibration, and electromagnetic interference), design deficiencies, manufacturing errors, maintenance errors, and operator errors.
> Personnel training; proper control room design; and operating, maintenance, and surveillance procedures are intended to afford protection from maintenance and operator errors.
> Additionally, provisions should be made to address common-cause failures. Examples of techniques are detailed defense-in-depth studies, failure mode and effects analysis, and analyses of abnormal conditions or events. Design techniques, such as diversity and defense-in-depth, can be used to address common-cause failures.

Based upon the IEEE 379-2000 guidance, the operator does not have to be assumed as a common mode failure. For this evaluation, the event consequences are unaffected by the operator common mode failure assumption but defense in depth (additional operators, procedures, time, caution tags) strategies all allow the operator to not be considered a common cause failure.

The accidents previously evaluated in the UFSAR were assessed for potential impacts and no adverse consequences were identified. The proposed change will ensure EFW will be available in a timely manner and capable of performing its specified function. 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:

This change addresses crediting a compensatory control room manual operator action to control Emergency Feedwater (EFW) backup flow control valves (EFW-223A(B)). The operator is performing the safety related function of controlling EFW flow via EFW-223A(B). The failure of the operator to perform this function either by omission or wrong action is considered an active single failure [ANSI/ANS-58.9-1981 Reference 7]. In general, design basis accidents have an event initiator and assume one active single failure. Each steam generator has two EFW flow paths available to provide sufficient

inventory to the steam generators. The design basis accidents have identified their limiting single failures with respect to a specific acceptance criteria, the potential operator action failure would be no different than the active single failure of one or both of the EFW backup flow control valves. With the failure of one or both EFW backup flow control valves, the EFW primary flow control valves would still be available to perform the EFW safety function.

UFSAR Table 10.4-14 (EMERGENCY FEEDWATER SYSTEM FAILURE MODE AND EFFECTS ANALYSIS) provides the failure modes and effects for the EFW system. The failure of the operator to perform the required action would be no different than the UFSAR Table 10.4-14 - failure of an EFS isolation valve to operate. The evaluation of this failure was that redundant valves and flow paths permit operation of the system and 200 percent pumping capacity remained available.

This change does not change any of the physical structures, systems, or components. The EFW system was designed to allow manual control of the EFW flow control valves. Some safety analyses currently assume operator control of EFW flow, typically after 30 minutes. UFSAR Section 10.4.9 already identified that after 10 hours manual remote EFW control was a required manual local operator action. The proposed change also does not create any new system interactions that could cause a malfunction.

Therefore, the operator action failure is no more adverse than that already analyzed in the UFSAR. The proposed change also does not create any new system interactions that could cause a malfunction. The proposed change does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

☐ Yes
☒ No

BASIS:

This change addresses crediting a compensatory control room manual operator action to control Emergency Feedwater (EFW) backup flow control valves (EFW-223A(B)). UFSAR Chapter 6 and 15 were reviewed to identify types of accidents that might be different than those previously evaluated.

UFSAR Chapter 15.1 (Increase in Heat Removal by the Secondary System) provides accidents in which the secondary side (Steam Generators) remove excess heat from the primary side (Reactor Coolant System). An inadvertent EFW actuation could add additional inventory to the Steam Generators (SGs). UFSAR Section 15.1.1.2 addresses this accident caused by the startup of emergency feedwater system. This change would have no impact on this event because it already assumes the EFW is actuated and cools down the secondary system.

The EFW system is primarily an accident mitigation system. The function and use of the EFW system will remain the same. The timing of the EFW system flow was evaluated and shown not to impact the accident consequences. The proposed change also does not create any new physical system interactions that could cause a different type of accident.

Thus, an accident of a different type than previously evaluated 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:
This proposed change does not impact the EFW physical configuration or the EFW physical system components. The only change is to place the EFW backup flow control valves in manual mode versus automatic which is already the design function and operation of the EFW system. 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.

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 [Reference 2], NRC REGULATORY ISSUE SUMMARY 2005-20 [Reference 3], and NRC Inspection Manual [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' UFSARs 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 an Unreviewed Safety Question (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 UFSAR, (2) create the possibility of an accident or a malfunction of a different type than any previously evaluated in the UFSAR, 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 is evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994. The ten primary attribute evaluations are specifically listed below.

(1) The specific operator actions required (procedures);

The specific operator action will be to control EFW-223A(B) in manual from the control room during a design basis event or LCP43 during a control room fire. Operator action to control EFW flow after 30 minutes is already typically assumed in the safety anlayses. LCP43 is only credited for a control room fire type event and is included for completeness. Item (4) below gives the specific procedures that will instruct this action. No adverse impact.

Note: The fire protection information is included in the 50.59 documentation to allow all evaluations to be included in one document. As stated in NEI 96-07 Section 4.5.1, applying 10CFR 50.59 to fire protection program changes is not required. Waterford 3 Operating License Section 2.C.9 states: EOI may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of fire. This evaluation demonstrates that the achieving and maintaining safe shutdown is not adversely impacted.

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

The operation of EFW-223A(B) will be from the control room or the remote shutdown panel (LCP43). Thus, potentially harsh or inhospitable environmental remain within the current design basis. No adverse impact.

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

The operation of EFW-223A(B) will be from the control room or LCP43. The LCP43 action is for fire related events only. The control room fire ingress/egress pathway has been validated in procedure OP-901-502 and OP-901-502-01. Thus, the ingress/egress paths are those within the current design basis. No adverse impact.

(4) The procedural guidance for required actions;

The operator actions required to control EFW during a design basis accidents are contained in the Emergency Operating Procedures. Immediate operator actions are contained in OP-902-000 and then optimal or functional emergency operating procedures are entered via procedures OP-902-001 through OP-902-008.

OP-902-000 (Standard Post Trip Actions) [Reference 21] provides the immediate operator actions. Step 6 provides a required Steam Generator level band and verification the EFW is available to restore inventory into the required band.

OP-902-001 (Reactor Trip Recovery Procedure) [Reference 22] is for an uncomplicated reactor trip with feedwater available. EFW is not needed in this optimal procedure. Step 9 provides the required Steam Generator level band.

OP-902-002 (Loss of Coolant Accident Recovery) [Reference 23] Step 31 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-003 (Loss of Offsite Power / Loss of Forced Circulation Recovery) [Reference 24] Step 10 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-004 (Excess Steam Demand Recovery) [Reference 25] first isolates the impacted Steam Generator then Step 16 uses EFW in manual to feed the unaffected Steam Generator. Step 22 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-005 (Station Blackout Recovery) [Reference 26] Step 10 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level. Step 17 provides monitoring for the nitrogen accumulators and provides instructions of manually operating the EFW flow control valves and the atmospheric dump valves.

OP-902-006 (Loss of Main Feedwater Recovery) [Reference 27]Step 11 gives options for the feedwater recovery. Step 13 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-007 (Steam Generator Tube Rupture Recovery) [Reference 28] Step 16 and 17 determines and isolates the most affected Steam Generator. Step 26 provides the required Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

OP-902-008 (Functional Recovery Procedure) [Reference 29] provides multiple success path which include a Steam Generator level band and instructs the use of EFW in auto or manual to maintain level.

The emergency operating procedures already contain the needed information to instruct the operator on the required actions. The emergency operating procedures would bound those actions prescribed in the off-normal procedures. OP-901-502 is discussed separately.

OP-901-502 (Evacuation of Control Room and Subsequent Plant Shutdown) [Reference 30] Attachment 3 (At the control time critical action) Step 4 already places EFW-223A(B) in manual to control steam generator level within a procedurally prescribed band. This

action is performed by the At the Control Operator and has been time validated in OP-901-502-01 (Time Critical Task Resource Management for Control Room Evacuation) [Reference 31]. This change is consistent with the fire safe shutdown analysis, thus no adverse impact.

IMC-326 [Reference 4] provides the following guidance:

Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. Accordingly, it is not appropriate to consider SSCs operable by taking credit for manual action in place of automatic action for protection of safety limits.

Technical Specification Section 2 provides the limiting safety system settings and safety limits. The EFW flow control valve setpoints are not TS Section 2 limiting safety system settings or safety limits, thus this change is not excluded by IMC-326 guidance.

In design basis space, the initiating event and a single active failure is all that is required to be evaluated. The EOP procedures provide appropriate direction for the operator to ensure this action is completed in the proper sequence. For beyond design basis scenarios with multiple failures, the OP-902-000 Standard Post Trip Actions (SPTAs) are organized around those critical safety functions which must be satisfied when a reactor trip is actuated or required, in order to ensure that the plant is placed in a stable, safe condition or that the plant is configured to further respond to a continuing casualty. Since, the EOPs are based upon critical safety functions, the highest priority functions would be performed earlier in the procedure than the lower priority functions. The EOPs are already organized appropriately to ensure the most important safety functions are met and the required operator actions are performed.

No adverse impact.

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

The EFW-223A(B) operator action will be performed from the control room or LCP43 in accordance with the operations procedures listed in Item (4). The operator will be required to be licensed and current on their license requirements. The operating procedures are those trained on during requalification cycles so no additional training is required. A licensed operator is already the minimum requirement to perform control board actions. In addition, an operations standing instruction will require each operator taking the assignment to be briefed on the compensatory measure the first time prior to standing watch. No adverse impact.

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

In order to ensure the action is completed during a design basis accident, an additional dedicated operator will be utilized for this action. During a design basis accident, multiple distractions can occur with only the Technical Specification minimum staffing available. To ensure this action is properly handled a dedicated operator will be assigned to this task. The dedicated operator's only emergency operating task will be to perform the EFW manual action. With being dedicated to only this task, there is an even higher assurance that the procedure actions will be accomplished as necessary than without a dedicated operator.

The OP-901-502 (Evacuation of Control Room and Subsequent Plant Shutdown) event due to a control room fire does not require the dedicated operator because the procedural guidance already has the At the Control operator place EFW-223A(B) in manual and control steam generator levels. This action is performed by the At the Control Operator and has been time validated in OP-901-502-01 (Time Critical Task Resource Management for Control Room Evacuation). This change is consistent with the fire safe shutdown analysis.

Dedicated operator requirements are as follows:

a. License operator current on their qualifications
b. Limited to the control room envelope
c. No concurrent duties during a design basis event (control room fire excluded)
d. Briefed on the compensatory measure (operations standing order) prior to the first time standing the dedicated watch

(e.g. An On Shift Field Operator that is licensed, limited to the control room envelope, and briefed would be allowed to perform the dedicated function because the field operator's only required function is during the control room/cable vault fire event [Reference 32].)

Technical Specification Table 6.2-1 minimum shift staffing requires two (2) reactor operators in Modes 1-4. Technical Specification 3.7.1.2 (Emergency Feedwater System) is applicable in Modes 1-3. As defense in depth, Technical Specification Table 6.2-1 also requires one (1) SM, (1) SRO, and (1) STA. The control room staff will assist the operator in validating EFW control.

(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 determination of whether operator action is required begins with the entry into the OP-902-000 standard post trip actions. The operator will use indications on control room panel CP-8 (or LCP43) which are control room design accident indications for steam generator level and emergency feedwater response. The steam generator levels restoring to the procedural bands will be indication of the current response. These parameters are also monitored by other members of the control room staff (control room supervisor, shift technical advisor, at the controls operator). 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;

ANSI/ANS-58.9-1981 [Reference 7] describes the requirements for single failures during design basis accidents. ANSI/ANS-58.9-1981 Section 3 rules states that a design basis accident plus single active failure is required to be analyzed. ANSI/ANS-58.9-1981 Section 3.7 states that an operator error is considered a single active failure.
ANSI/ANS-58.8-1994 Section 3.1.3 states safety-related operator actions shall be credited only where a single operator error of omission does not result in exceeding any limiting design requirement for the DBE under consideration.

The ability to recover from creditable errors remains within the bounding single failures already evaluated in the accident analyses. The timing to identify a wrong operator action would be consistent with the operator timing to validate critical safety functions are

met in OP-902-000 (Standard Post Trip Actions). CR-WF3-2011-1714 CA#30 was used to determine the time for all crews to complete OP-902-000. All crews consistently completed OP-902-000 actions in less than 10 minutes during a Station Blackout timed evaluation. In addition, during the CR-WF3-2015-3565 event, operations took manual control of EFW-223A(B) within approximately 5 minutes of the EFW flow oscillations start (based upon PI data).

As defense in depth, a caution tag is being placed on each control room EFW-223A(B) switches to signify the impacted valves. In addition to the dedicated operator, the balance of plant operator, the control room supervisor, shift technical advisor, and at the controls operator will also be validating plant conditions and are able to identify any credible error. CR-WF3-2011-1714 CA#30 has validated that the operator can complete OP-902-000 standard post trip actions within 10 minutes.

No adverse impact.

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

Considering the availability of cues to the operators, the non-adverse environment, locality of the action, available procedures, and qualified personnel, the impact of this change on plant risk is expected to be negligible. The risk significance for EFW initiation has not changed. The EFW will still actuate when required and the operator action ensures sufficient feeding to maintain steam generator desired levels. No adverse impact.

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

NRC INFORMATION NOTICE 97-78 and NUREG-0800 APPENDIX 18-A both state that the use of ANSI/ANS-58.8-1994 is appropriate for determining safety related manual operator actions times. NUREG-0800 APPENDIX 18-A is for similar evaluations of digital control system manual action credit.

NRC INFORMATION NOTICE 97-78 states:

> ANSI-58.8 provides estimates of reasonable response times for operator actions; however licensees may use time intervals derived from independent sources provided they are based on analyses with consideration given to human performance.

NUREG-0800 APPENDIX 18-A (CREDITING MANUAL OPERATOR ACTIONS IN DIVERSITY AND DEFENSE-IN-DEPTH ANALYSES) [Reference 20] states that acceptable methods for deriving analysis time estimates for individual task components include, but are not limited to:

- Use of control/display mockups
- ANSI/ANS 58.8, "Time Response Design Criteria for Safety-Related Operator Actions"

This section will first address nitrogen volume available and associated time the operator would have to take action. Then, the ANSI/ANS-58.8-1994 information will be used to ensure sufficient time is available based upon this time information.

EC58184 demonstrates that allowing up to 30 minutes for operator action is acceptable in terms of nitrogen supply volume. EC58184 describes that the current design basis for the nitrogen accumulator accounts for twenty (20) EFW-

223A(B) strokes. At 30 minutes, EC58184 Attachment 1 showed that at least 11 full EFW-223A(B) strokes remained available for the credited operator action. The EC58184 limiting case considers a failure of the EFW primary flow control valve with a secondary side break on the opposite steam generator as the limiting scenario. The EFW backup flow control valves, EFW-223A(B) are in manual control and are not manually operated. At 36.3% wide range (WR), a priority open signal is generated that overrides manual control and fully opens the primary and backup flow control valves to supply maximum flow to the unaffected steam generator. The priority open signal resets at 40% WR and the EFW control valves return to their previous setting (closed). This repetition occurs until the operator action is credited to take control of EFW flow. The analysis determined that at 11 minutes 16 out of 20 closed strokes would remain with conservative steaming and EFW flow rates; at 30 minutes 11 out of 20 closed strokes would remain. The EFW flow control valves are fail open valves so the open stroke (spring force) does not use nitrogen volume only the closed stroke. The remaining 11 valve strokes is sufficient because the operator would set the EFW backup flow control valve to a throttled position (flow value) and then make minor adjustments to stabilize steam generator level within the procedural requirements. This is consistent with the EC58184 assumptions.

ANSI/ANS-58.8-1994 provides time requirements for different accident scenarios. ANSI/ANS-58.8-1994 Section 4.1 Table 1 shows the following for different plant conditions.
Plant Condition 2 – Operator Diagnosis Time is 5 minutes
Plant Condition 3 – Operator Diagnosis Time is 10 minutes
Plant Condition 4 and 5 – Operator Diagnosis Time is 20 minutes

ANSI/ANS-58.8-1994 Section 4.1 Table 2 shows the following for operator action times.
Plant Condition 2 – Operator Action Time is 1 minute + n
Plant Condition 3 – Operator Action Time is 3 minutes + n
Plant Condition 4 and 5 – Operator Action Time is 5 minutes + n
n signifies the number of discrete actions.

ANSI/ANS-58.8-1994 allows for the worst plant conditions (4 and 5) a diagnosis time of 20 minutes and then 5 minutes + n for each action. If one action is to increase demand on EFW-223A and another action is to increase demand on EFW-223B, the total time would be 20 minutes + 5 minutes + 2 minutes = 27 minutes. This is within the EC58184 time of 30 minutes with 11 full EFW-223A(B) strokes available, thus the operator action time of up to 30 minute is acceptable. The UFSAR Chapter 15 events were demonstrated to have no adverse impact with manual control of the backup flow control valves for the first 30 minutes post-accident. 30 minutes is also listed in ANSI/ANS-58.8-1 for operator actions required outside the control room which is not required for this condition. The dedicated operator will be stationed in the control room envelope to perform the required procedural actions. The Emergency Operating Procedure (EOP) actions correspond to ANSI/ANS-58.8-1994 diagnosis actions required to identify the event and determine the required actions.

As confirmation that 30 minutes operator action time is acceptable, CR-WF3-2011-1714 CA#30 was used to determine the time for all crews to complete OP-902-000 (Standard Post Trip Actions). This empirical human performance data showed that all crews consistently completed OP-902-000 actions in less than 10 minutes during a Station

Blackout timed evaluation. This provides assurance that during an actual event, these actions could be accomplished within the 30 minute time frame. The station blackout timed evaluation would be similar in terms of operator requirements for any of the UFSAR Chapter 15 events. There is even higher assurance that procedural actions for EFW flow control will be completed by a dedicated operator with no other distractions or duties.

For beyond design basis events, any additional failures could take additional time but that would just use the time margin between the OP-902-000 (Standard Post Trip Actions) 10 minutes and the 30 minutes needed to control the EFW backup flow control valves. In addition, the EOPs step for maintaining steam generator level is "*" which means this step is continually performed and may be pulled earlier in the procedure if required. In addition, during the CR-WF3-2015-3565 event, operations took manual control of EFW-223A(B) within approximately 5 minutes of the EFW flow oscillations start (based upon PI data).

The OP-901-502 (Evacuation of Control Room and Subsequent Plant Shutdown) event due to a control room fire does not require the dedicated operator because the procedural guidance already has the At the Control operator place EFW-223A(B) in manual and control steam generator levels. This action is performed by the At the Control Operator and has been time validated in OP-901-502-01 (Time Critical Task Resource Management for Control Room Evacuation). This change is consistent with the fire safe shutdown analysis. For the fire analysis, the time validation and procedural guidance already exists for the at the controls operator to place EFW control in manual and control steam generator levels. OP-901-502-01 shows that the required action can be completed within 25 minutes which is within the EC58184 time of 30 minutes which ensures 11 out of the 20 EFW valve strokes remain available. This change does not impact those procedural actions, thus this evaluation is documenting that the control room fire event already bounds the EFW manual control.

NOTE: The 10CFR50.59 Revision 0 evaluation credited 11 minutes for the dedicated operator action. The 11 minutes is still a valid time but this evaluation uses the more conservative 30 minutes. ANSI/ANS-58.8-1994 Section 5.1 (Use of Performance Data) states that in lieu of the requirements specified in Section 4 Tables 1 and 2, the designer may use time intervals derived from empirical human performance data if a 95% confidence level is demonstrated for the time available to perform safety-related operator actions. CR-WF3-2011-1714 CA#30 was used to determine the time for all crews to complete OP-902-000 (Standard Post Trip Actions). All crews (100%) consistently completed OP-902-000 actions in less than 10 minutes during a Station Blackout timed evaluation. It is a reasonable expectation that a dedicated operator with no other distractions or duties would be able to perform the intended function faster than the 10 minute time demonstrated in CR-WF3-2011-1714 CA#30. The intent of using the dedicated operator was to ensure the action timing would have a higher assurance of completion than that identified in CR-WF3-2011-1714 CA#30. Thus, the previous use of the 11 minutes is still considered conservative and appropriate.

In conclusion, ANSI/ANS-58.8-1994 was used to determine the operator action time within 30 minutes is allowable. As defense in depth, EC58184 calculated that 69.9 minutes is available prior to all 20 EFW backup control valve strokes being used with the design basis volumes. This means margin exists between that credited in this evaluation and the time when the nitrogen accumulator volume reserved for the EFW backup control valves will be

consumed. The nitrogen accumulator sizing calculation (ECM88-024) shows that only 1% of the stored nitrogen is dedicated for stroking EFW-223A(B) and more than 50% is reserved for potential leakage. The most recent surveillance test demonstrated that actual leakage is less than half the allowed leakage. With this available nitrogen accumulator margin, it would take over 6 hours prior to the accumulator depletion. Thus, significant nitrogen accumulator margin exists that could be used during design basis accident.

Since, the regulatory requirements have been met to credit the operator action, this 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:

This change addresses crediting a compensatory control room manual operator action to control Emergency Feedwater (EFW) backup flow control valves (EFW-223A(B)). This could impact the amount of water and timing of inventory to the steam generators but does not specifically cause the fuel clad, reactor coolant system boundary, or containment to fail. The steam generator inventory does impact the amount of scrubbing the water performs in the radiological analysis which is pertinent to the effectiveness of the fission product barrier. The scrubbing is described as a partition factor (PF) which is used to calculate what fraction of non-gaseous releases remain in the water and what fraction is released to the environment.

Calculation CN-TAS-03-30 [Reference 14] is the AOR for Nuclear Steam Supply System (NSSS) response for the radiological analyses. CN-TAS-03-30 Appendix I provides a justification for the steam generator partition factors used in the radiological analyses. When the steam generator tubes are covered, a reduction in the amount of radiological releases is allowed since any leak will be submerged so scrubbing of that release is credited. This is reflected by a change in the partition factor used in the radiological dose release calculation.

CN-TAS-03-30 Appendix I Section I.3 (Symmetric Steam Generator Non-Depletion Event) covers CEA Ejection, Sheared Shaft/Seized Rotor, SBLOCA, Letdown Line Break, and Excess Load with LOAC events. Symmetric steam generator non-depletion event means all the radiological events which will have two (2) steam generators available for heat removal with no secondary side accident. This section assumed an operator action at 30 minutes to establish EFW flow (575 gpm). Prior to operator action, EFW is assumed to be in automatic. This scenario requires 6113 seconds for the steam generator tubes to be covered. Figure I.3-1 shows prior to 1800 seconds, EFW was 27.7 lbm/sec (200 gpm) which is equivalent to the flow through the EFW primary flow control valves. For steam generator tube submergence, the limiting single failure is the failure of half of the EFW flow which only leaves 575 gpm available. This information in Section I.3 is consistent with the assumed manual action at 30 minutes to control the EFW backup flow control valves (EFW-223A(B)), so these radiological events remain valid.

CN-TAS-03-30 Appendix I Section I.4 (Single Steam Generator Depleting Event) covers the feedwater line break event. This section assumed operator action at 30 minutes to establish EFW flow (575 gpm). Prior to operator action EFW is assumed to

be in automatic. For steam generator tube submergence, the limiting single failure is the failure of half of the EFW flow which only leaves 575 gpm available. This scenario requires 6695 seconds for the steam generator tubes to be covered. The releases calculated in the main body and used for Alternate Source Term (AST) were done assuming that a PF of 1 existed for the first 14,400 seconds (4 hours). A PF of 1 means no partitioning or scrubbing for that time period (that is, it assumes the steam generator tubes are not covered until 14,400 seconds). This extended time when no scrubbing of the radioactive release is credited conservatively bounds any impact of automatic EFW flow assumed within the first 30 minutes. Operator manual control of EFW flow at 30 minutes ensures that the tubes are covered within 14,400 seconds, In addition, 14,400 seconds exceeds 6695 seconds by more than 1800 seconds (operation action credit time), so there would be no impact to this analysis for the assumed operator action.

CN-TAS-03-30 Appendix I Section I.5 (Single Steam Generator Non-Depleting Event) covers the steam line break and steam generator tube rupture events. This section assumed operator action at 30 minutes to establish EFW flow (575 gpm). Prior to operator action EFW is assumed to be in automatic. For steam generator tube submergence, the limiting single failure is the failure of half of the EFW flow which only leaves 575 gpm available. This scenario requires 3516 seconds for the steam generator tubes to be covered. For the first 30 minutes of the event, the partition factor assumed is 1 (all radioactive leakage into the steam generator is released for the first 30 minutes) and then 100 for the remaining time to account for the reduced release caused by flashing of hot RCS water as it leaks into the steam generator. Figure I.5-1 shows the fraction of leakage which flashes to steam after 1800 seconds is less than 0.01, thus the associated partition factor would be greater than 100 (1 / 0.01 = 100). These assumptions make EFW flow to cover the steam generator tubes irrelevant to the radiological dose analysis (since EFW flow is prevented from going to the broken steam line break generator). Operator manual control of EFW flow at 30 minutes ensures there is adequate heat removal from the intact steam generator consistent with the analysis. Therefore, there would be no impact to these radiological does release analyses for the assumed operator action.

For the potential to overfill the steam generator due to operator action which could cause a failure of the main steam line piping, calculation ECS02-005 (Effects of Uncontrolled EFW on Accident Situations) [Reference 17] already credits operator action to either stop EFW flow or close the impacted steam generator isolation valves when the high steam generator level alarm is actuated. This change will not impact this event.

With credit of the operator action, the steam generator water level and feed rates will be within the current design basis requirements. Thus, the radiological consequences will not be impacted. This ensures that the current limiting safety analysis for dose consequences and fission product barrier limits remain bounding. 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:

Methods of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC [Reference 6 Section 3.10]. This change is not impacting the methodology or topical reports used in the UFSAR analyses. The existing UFSAR evaluations utilizing EFW as a mitigating system have 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.

I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 (WF3) **Evaluation # 2015-02 / Rev. #:** 0

Proposed Change / Document: Fukushima EP Communications – EC 47846 & LBDCR 15-019 **Rev. #:** 0

Description of Change:

Engineering change (EC 47846) provides permanently installed sources of 120 Vac uninterruptible power supplies (UPSs) power to communications equipment required to be maintained for 24 hours after the onset of a Beyond Design Basis External Event (BDBEE) and additional Emergency Planning (EP) radios and satellite phones as required for compliance with NEI 12-01. After the initial minimum 24 hour period, lighting panels being re-energized by the FLEX Diesel Generator (installed by EC 48145) and power cords will provide continuous power to the credited equipment in the RAB.

Although the scope of EC 47846 includes changes in several areas, the affected areas (relevant to the licensing basis) are the Technical Support Center (TSC) and the Reactor Auxiliary Building (RAB) +7' Work Area. EC 47846 installs Valve-Regulated Lead-Acid (VLRA) batteries in these areas. This type of battery has the potential to generate small amounts of hydrogen gas (H_2). The generation of H_2 was "screened in" by the Process Applicability Determination (PAD) because a loss of ventilation in the affected areas may lead to an unacceptable H_2 accumulation under design basis conditions. As a result of this possibility for hydrogen accumulation, the RAB Air Conditioning System and the Control Room Ventilation System have a new design function to prevent the accumulation of hydrogen gas.

Hydrogen gas has a Lower Explosive Limit (LEL) of 4% when mixed with a normal atmosphere. IEEE 1187-2002 (IEEE Recommended Practice for Installation Design and Installation of Valve-Regulated Lead-Acid Storage Batteries for Stationary Applications) specifies a hydrogen accumulation limit of 2% of the total space volume to provide margin to the LEL. The basis for the acceptability of this change is that the design ensures that hydrogen accumulation will remain below the threshold value of 2%.

The condition that could create excessive hydrogen accumulation is a loss of ventilation while the batteries are still charging.

Technical Support Center

EC 47846 Installs one (1) new UPS with one (1) external battery pack inside the TSC to provide 24 hours of operation to the radio deskset in the TSC, the two radio desksets in the Main Control Room (MCR), and a permanent satellite phone docking station in the RAB +69' Elevation stairwell.

The TSC is contained with the Control Room Envelope (CRE) which is defined in UFSAR 6.4.2.1:

> *The control room envelope is defined to include the main control room, computer room, computer room air conditioning equipment room, control room HVAC equipment room, emergency living quarters* [also known as the TSC], *emergency food and water storage rooms, toilets, locker rooms, kitchen, kitchenette, supervisors office, corridors, conference room and vault (critical document reference file).*

The TSC is ventilated by the Control Room Air Conditioning System [UFSAR 9.4.1(f)]. The design basis function applicable to this evaluation is to [UFSAR 9.4.1.1(c)]

> *permit personnel occupancy and proper functioning of instrumentation and controls during all normal and design basis accident conditions assuming a single active failure coincident with a loss of offsite power...*

The new design function of limiting hydrogen concentrations below 2% for all operating conditions is demonstrated by the response to Question 2 and calculation 5-F.

[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.

RAB +7' Work Area

EC 47846 installs two (2) UPSs and ten (10) battery packs in RAB +7' Elevation Work Area (also referred to as "Office Area," "Administrative Area," or "Old Health Physics Area"), to provide 24 hours of operation to the Maintenance and Operations radio channels.

This area is ventilated by the RAB Air Conditioning System[2]. UFSAR 9.4.3.7.1 states:

> *The RAB Air Conditioning System is not required to prevent or mitigate the consequences of a design basis accident or to provide a safe shutdown of the reactor. Therefore, the system is non-safety and nonseismic.*

There are two battery chargers: one powered by a Safety-Related panel (PDP-387A) and the other by diesel generator-backed security panel. Each panel also powers an air mixer, also installed by EC 47846, which is designed to prevent hydrogen gas pocketing or stratification. Any design basis accident with coinciding Loss-of-Offsite-Power (LOOP) will leave the area without ventilation with the batteries still capable of hydrogen production. The air mixer powered by the Safety-Related panel allows a "time-to-2%" calculation (5L2) to assume that the hydrogen is well mixed with the air in the room.

A review of PDMS indicates that the RAB +7' Work Area contains Safety-Related power cables:

- Cable 31531A in conduit 31531A-SA is routed from 4kV SWGR 3A3S to Emergency Feedwater Motor Driven Pump A (EFW-MPMP-0001A)
- Cable 32531A in conduit 32531A-SAB is routed from 480V SWGR 3AB31S to 480V MCC 3AB311S (SSD-EMCC-311AB)
- Cable 32531B in conduit 32531B-SAB is also routed from 480V SWGR 3AB31S to 480V MCC 3AB311S (SSD-EMCC-311AB)

MCC 3AB311S feeds Emergency Feedwater Pump Turbine Stop Valve, Charging Pump AB Cooler AH-22, Charging Pump AB Cooler AH-23, Battery Charger 3AB1-S, Battery Charger 3AB2-S, Aux Building Chiller Water Pump P-1, Component Cooling Water Pump AB Cooler AH-20, SUPS 3AB, and Safeguard Pump Room AB Cooler AH-21.

The evaluation provided by calculation 5L2 shows that these power cables and the associated systems are not subject to compromise because the circumstances which could lead to a hydrogen explosion will not occur. The affected operating procedures (OP-500-002, OP-100-014 and OI-004-000) will be revised as required to have Operations initiate a corrective action when AH-5 and/or associated mixing fans in the Work Area are inoperative. Appropriate procedures will be revised (AR 00229188) to have support personnel implement corrective actions for any off-normal event which prevents air removal/recirculation during design basis events. These procedure changes will ensure that corrective actions will be performed if AH-5 is not available to perform its new design basis function of hydrogen accumulation prevention during normal operations and DBA conditions.

[2] Specifically, the area is ventilated by air handling unit HVRMAHU0017 / HVRMFAN0018 (AH-5(3)), hereafter referred to as AH-5.

Summary of Evaluation:

For all design/licensing basis conditions, there are no circumstances which could lead up to a hydrogen explosion. The TSC is ventilated at all times except for the isolation mode for which the batteries cannot produce enough hydrogen to reach 2% because the ventilating system recirculates the control room envelope air, allowing the entire volume of the CRE to dilute the hydrogen.

The rate at which hydrogen could accumulate in RAB +7' Work Area is too low to be considered within any Design Basis Accident timeframe, and procedures will be revised to ensure that the Work Area is adequately ventilated (to remove or dilute hydrogen) for any condition which would have AH-5 shut down for any appreciable length of time. Therefore, the evaluation concludes that the proposed change does not require prior NRC approval for implementation.

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: F. Bivins Calhoun III / See AS / ENERCON / DE-Mechanical / 6/23/15
Name (print) / Signature / Company / Department / Date

Reviewer: Sada Dhingra / See AS / Enercon / DE-Mechanical / 6/23/15
Name (print) / Signature / Company / Department / Date

OSRC: Ran Gilmore / Ran B Gilmore / 7-1-15
Chairman's Name (print) / Signature / Date

W3 15-11
OSRC Meeting #

II.	**50.59 EVALUATION** **Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.**	☐ ☒	Yes No
Does the proposed Change:			
1.	Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?	☐ ☒	Yes No
	BASIS: Hydrogen accumulation in the affected areas (TSC and RAB +7' Work Area) is not the initiator of any accident previously evaluated in the UFSAR, nor does any previously evaluated accident have an origin in the affected areas. In addition, a loss of ventilation by any UFSAR-described system is not the initiator of any accident previously evaluated in the UFSAR. The UFSAR does not address the subject of H_2 accumulation anywhere in the RAB (with the exception of the battery rooms [UFSAR 9.4.3.5.1(b)] and the Heating & Ventilation Equipment Room [UFSAR 9.4.3.4.2]), including the Control Room Envelope (CRE). The safety related equipment associated with the affected areas (see Section I) are not considered to be accident initiators. Therefore, the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.		
2.	Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR?	☐ ☒	Yes No
	BASIS: The new design functions for the RAB Air Conditioning System and the Control Room Air Conditioning System is to limit the accumulation of hydrogen to a level below 2%. Hydrogen Production This function is accomplished in the TSC since it is continually ventilated during all normal and design basis accident conditions, except for a Station Blackout (SBO) event which is limited to four hours (UFSAR Appendix 8.1A, Station Blackout Evaluation). This four hour event is much too short for hydrogen to accumulate to an appreciable amount (see below for time to 2%). The TSC will not lose ventilation due to a Design Basis Accident (DBA) coincident with Loss of Offsite Power (LOOP) since the Control Room Air Conditioning System (which ventilates the TSC) is provided with redundant power sources as discussed by UFSAR Section 9.4.1: *Permit personnel occupancy and proper functioning of instrumentation and controls during all normal and design basis accident conditions assuming a single active failure coincident with a loss of offsite power* UFSAR Section 6.4.3.3 states that no outside air is drawn into the control room envelope during a toxic chemical emergency. During this mode of operation the air in the CRE is circulated, allowing the entire CRE to dilute the hydrogen supply. Calculation 5-F indicates that the entire life cycle of the batteries will not produce enough hydrogen to reach the threshold of 2%. Therefore, the Control Room Air Conditioning System can accomplish the new design function of limiting the accumulation of hydrogen to a level below 2%. A new non-safety related EP UPS which contains valve regulated lead acid (VRLA) batteries is being installed in a room served by non-safety related AH-5. This room contains Division 1 Class 1E circuits relied upon to safely shut down the unit following a DBA. AH-5 is being credited to minimize the accumulation of a combustible concentration of hydrogen generated by the EP UPS in the room during normal plant conditions. During normal plant operation, the EP UPS is maintained charged via connection to Division 1 Class 1E SWGR 3A31-S. During a Loss of Offsite Power Event (with or without an accompanying DBA), the UPS is maintained charged via Division 1 Class 1E SWGR 3A31-S or SWGR 3AB31-S powered by an EDG, however, non-		

safety related air handling unit 5 is not powered during a Loss of Offsite Power since it is not powered from a EDG backed MCC.

The impact of hydrogen generation during times when the UPS is being charged with no HVAC available was evaluated. It was determined by the evaluation that, with no mitigating actions taken (e.g., opening of doors or providing supplemental ventilation) or the restoration of the HVAC system, accumulation of hydrogen for in excess of six to eight months would be required to reach the WF3 specified limit of 2% room volume concentration due to the extremely small amount of hydrogen generated by the UPS. Therefore, during any event that results in the loss of AH-5, the low hydrogen generation rate inherent to VRLA batteries during charging and the adequacy of the room volume (with at least one air mixing fan) is the primary means of limiting hydrogen accumulation.

Operations Shift Logs will require the initiation of corrective actions to track out of service time for AH-5 and provide any actions required to ensure continued safe operation of the plant (for example, opening of doors and/or supplemental ventilation). Should the loss of ventilation (i.e., loss of AH-5) occur during a Loss of Offsite Power (with or without a DBA) when the UPS is being charged, revised plant procedures will provide the necessary guidance to restore hydrogen removal capability in a timely manner (six to eight months per evaluations performed). Following restoration of offsite power and restoration of the balance of plant power distribution system, AH-5 could be restored to operation if available or actions (e.g., utilization of supplemental ventilation) could be taken to continue to maintain the hydrogen level well below the WF3 specified limit of 2% room volume concentration. With this extremely long window (six to eight months) with no challenge to SSCs important to safety available, significant resources (both personnel and equipment) will be available to address the loss of ventilation in the room.

Therefore, the activity does not result in more than a minimal increase in the likelihood of a malfunction of an SSC important to safety since the design of the installation, room volume, the inherent characteristics of the VRLA batteries, and significant time for plant response for any required mitigating actions provide defense in depth for ensuring that hydrogen concentration is maintained well below the WF3 site limit of 2% room volume which ensures the continued availability of the Class 1E circuits contained within the room.

Battery Failure

Although rare, industry experience indicates that VLRA batteries have a failure mode due to thermal runaway. Thermal runaway occurs when more heat is generated within the battery than can be dissipated through the battery casing. This failure mode is typically associated with high charging voltage and high ambient temperature, in a positive feedback loop of increasing voltage and temperature until the battery fails by exploding and/or burning. This type of failure can result in the generation of hydrogen sulfide and other toxic fumes created by combustion of the plastic battery casing.

Technical Specification Surveillance Requirement 4.7.6.1(b) requires that the Control Room air filtration HEPA filters are to be demonstrated as Operable following painting, fire, or chemical release in any ventilation zone communicating with the system. This requirement indicates that filtration train failure is an anticipated event.

There are multiple factors which demonstrate that a toxic gas event degrading the charcoal filters is not a credible event. The Control Room Envelope is a controlled environment with the ambient temperature maintained between 70° to 75° (UFSAR Table 9.4-1), the UPS has built-in over-current and over-voltage protection and, should the temperature get too high (121°F) the UPS (which right next to the batteries) will shut down to prevent damage to the electronics and batteries.

Conclusion

The proposed change does not cause a greater reliance to be placed on any specific system, structure or component to perform a safety function. The ventilation systems are not required to operate under different conditions than those stated by the UFSAR to prevent an accumulation of hydrogen. Control Room air filtration train damage due to battery failure is an unlikely event

	which is protected against by Technical Specification Surveillance Requirements. The proposed change does not degrade the safety function of any safety related equipment or equipment important to safety which might mitigate malfunction of equipment. Therefore, the proposed activities will not increase 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 addition of the design function for the RAB Air Conditioning System and the Control Room Air Conditioning System to limit the accumulation of hydrogen to a level below 2% does not change, prevent or degrade the effectiveness of actions described or assumed in an accident discussed in the UFSAR, nor does it affect the function of equipment designed to control the release of radioactive material. The affected systems (RAB and Control Room Air Conditioning) will not be operated in a different manner in response to any radiological event. The proposed activity does not result in a new pathway for release of radioactive material or affect onsite dose in a way that restricts access to vital areas or impedes mitigating actions. Although rare, industry experience indicates that VLRA batteries have a failure mode due to thermal runaway. Thermal runaway occurs when more heat is generated within the battery than can be dissipated through the battery casing. This failure mode is typically associated with high charging voltage and high ambient temperature, in a positive feedback loop of increasing voltage and temperature until the battery fails by exploding and/or burning. This type of failure can result in the generation of hydrogen sulfide and other toxic fumes created by combustion of the plastic battery casing. Technical Specification Surveillance Requirement 4.7.6.1(b) requires that the Control Room air filtration HEPA filters are to be demonstrated as Operable following painting, fire, or chemical release in any ventilation zone communicating with the system. This requirement suggests that a VLRA battery failure due to thermal runaway could damage the filters which are required to limit Control Room dose during a DBA. There are multiple factors which demonstrate that a toxic gas event degrading the charcoal filters during a DBA is not a credible scenario, not the least of which is that the toxic gas production is the result of a fire due to thermal runaway and fire events are not required to be postulated as occurring during a DBA. In addition, the Control Room Envelope is a controlled environment[3] with the ambient temperature maintained between 70° to 75° (UFSAR Table 9.4-1), the UPS has built-in over-current and over-voltage protection and, should the temperature get too high (121°F) the UPS (which is right next to the batteries) will shut down to prevent damage to the electronics and batteries. Therefore, the proposed activity will not result in an increase in the consequences of any 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: UFSAR Section 6.4.1(d) states that the functional design of the control room habitability systems are based on: *The radiation exposure to main control room personnel, throughout the duration of any one of the postulated accidents discussed in Chapter 15, does not exceed the limits of General Design Criterion 19 of Appendix A to 10CFR50 and 10CFR50.67.*		

[3] With the exception of a Station Blackout (UFSAR Appendix 8.1A). During the 4 hour SBO, the TSC temperature can be expected to increase due to loss of ventilation. However, the batteries will not be charging due to loss of all A/C power.

	Furthermore, UFSAR Section 9.4.1.1(a) states that the Control Room Air Conditioning System is designed to: *limit control room doses due to airborne activity to within General Design Criterion 19 (Appendix A of 10CFR50) and 10CFR50.67 limits...* The Control Room Air Conditioning System will continue to perform all its existing design functions. No changes are being made to the way the control room operators respond or to the way in which control room operators will take actions in the event of an abnormal conditions in the Control Room Air Conditioning System. Implementation of EC 47846 for installation of VRLA batteries will not result, either directly or indirectly, in an adverse effect to any safety functions required for analyzed accidents and do not increase any radiological hazards. The control room habitability systems are not affected and do not increase any radiological hazards. The RAB Air Conditioning System is not credited with mitigating the consequences of a malfunction of an SSC previously evaluated in the UFSAR. It has been demonstrated that hydrogen cannot accumulate to explosive levels in the RAB +7' Work room, so there is no risk to any nearby system which may be credited with mitigating the consequences of a malfunction of an SSC previously evaluated in the UFSAR. This new design function (hydrogen accumulation prevention) does not require the Control Room Air Conditioning System or the RAB Air Conditioning System to be operated any differently than they are now operated, or to be modified in any way. Therefore, no greater reliance is placed on either system to perform a safety function. Therefore, the proposed change will not result in more than a minimal increase in the consequences of a malfunction of SSC 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: EC 47846 will install Valve-Regulated Lead–Acid (VRLA) batteries in the TSC. These batteries recombine hydrogen gas, but can emit small amounts of hydrogen gas. As calculated in Attachment G of Calculation 5-F, the maximum hydrogen generation rate in the TSC during <u>normal operations</u> (i.e. the UPS and its associated battery pack on float charge) is 0.008694 ft^3/hr (0.000145 cfm). During a postulated <u>worst case</u> scenario (with respect to hydrogen generation rate) whereby a set of 2 batteries in the charging circuit and the UPS Battery fails and overcharges, the total hydrogen generation (Q_{Total}) in the TSC could only reach a maximum of 31.53 ft^3. Since this is less than the maximum allowable volume of hydrogen accumulation of 57.6 ft^3, hydrogen generation is not a concern during the postulated catastrophic failure scenario. Calculation 5-F, Control Room / Computer Area HVAC Cooling Loads determines the maximum hydrogen generation rate in the TSC during normal operations (float charge). Under ideal conditions (no ventilation and air-tight room), it would take approximately 276 days to reach the hydrogen accumulation limit of 2% of the room air volume specified by IEEE 1187-2002 for hydrogen accumulation. However, the TSC is continually ventilated during all normal and design basis accident conditions. The only exception is the control room isolation mode of operation. For this condition, the batteries during their entire lifecycle cannot produce enough hydrogen to reach 2% because the ventilation system recirculates the control room envelope air, allowing the entire volume of the CRE to dilute the hydrogen. EC 47846 will also install VRLA batteries in the RAB +7' Elevation Work Area. These batteries recombine hydrogen gas, but can also emit small amounts of hydrogen gas. As calculated in Attachment 1 of Calculation 5L2, the maximum hydrogen generation rate in the RAB +7' Elevation Work Area is 0.0753 ft^3/hr (0.0013 cfm) during <u>normal operations</u> (i.e. both UPS and all associated battery packs on float charge). This is considered negligible, as under these conditions, it would take approximately 242 days to reach the 2% limit for hydrogen accumulation in the Work Area		

on Elevation +7 of the RAB with no air exchanges / ventilation.

During the postulated worst case scenario (with respect to hydrogen generation) whereby a set of 6 batteries in the charging circuit (for each UPS) and the UPS Battery fails and overcharges, the total hydrogen generation (Q_{Total}) in the RAB +7' Elevation could only reach a maximum of 147.14 ft^3. Since this is less than the maximum allowable volume of hydrogen accumulation (i.e. 2% of the total room volume) of 438.8 ft^3, Hydrogen generation is not a concern during the postulated catastrophic failure scenario

Calculation 5L2, Office Area – HVAC Cooling Loads determines the maximum hydrogen generation rate in the RAB +7' Elevation Work Area during normal operations. This area is ventilated and there will be not be any hydrogen accumulation during normal plant conditions. A LOOP would shut down normal ventilation which is not powered by redundant, safety-related sources. However, the LOOP would not stop the VLRA batteries from potentially creating hydrogen gas. Under these conditions (no ventilation with batteries emitting hydrogen), it would take approximately 242 days to reach the hydrogen accumulation limit of 2% of the room air volume. In addition, a fan powered by the same circuit as the UPS which is charging the batteries will be circulating and mixing air in the area. Based on these conditions it is concluded that H2 will not accumulate in any measureable amount.

The Work Area contains Safety Related power cables which could possibly fail in the event of a hydrogen gas explosion – leading to 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. However, this evaluation demonstrates that circumstances that lead up to an explosion will not occur. In addition, the area contains multiple non-Safety Related cables – the failure of which could lead to plant transients or shutdown. However, their failure would not lead to an accident of a difference type since the plant is analyzed for safe shutdown with the absence of any non-Safety Related system.

The proposed changes include a determination during performance of the Operator Shift Logs that AH-5 and the air mixers[4] are operating and if they are not operating, to create a corrective action to restore them within an appropriate time period.. This will allow AH-5 to be relied upon to prevent hydrogen accumulation in the +7' Office Area during normal plant operations.

The action times are provided by calculation 5L2, and summarized in the Topic Notes:

The functionality of the Reactor Auxiliary Building (RAB) Air Conditioning System Administrative Area ventilation ensures that the accumulation of hydrogen generated from operation of the EP Communications UPS batteries in the RAB EL. +7' Elevation Administrative Area is minimized. It was determined by evaluation that if Administrative Area air handling unit HVRMAHU0017 / HVRMFAN0018 (AH-5(3)) becomes non-functional with air mixers (HVRMFAN0070-A and HVRMFAN0070-B) operational and no mitigating actions are taken, it would require a minimum of six months to accumulate enough hydrogen to reach the WF3 specified limit of 2% room volume concentration due to the small amount of hydrogen generated by the UPS during charging assuming uniform dispersion of hydrogen within the room. The operation of the Reactor Auxiliary Building (RAB) Air Conditioning System Administrative Area air mixers ensures that the air space is uniformly mixed and pocketing of hydrogen generated from the charging of the EP Communications UPS is actively prevented. The Administrative Area air mixers are credited when the Administrative Area air handling unit AH-5 is not operating; AH-5 is operating, the mixers provide supplemental mixing of the room's environment. It was determined by evaluation that with the Administrative Area air handling unit AH-5 and both Administrative Area air mixers out of service, and no mitigating actions taken, it would require approximately 30 days to accumulate enough hydrogen to reach the WF3 specified limit of 2% room volume concentration due to the small amount of hydrogen generated by the UPS during charging. An allowed outage time of 60 days for the Administrative Area air handling unit AH-5 with both Administrative Area air mixers functional is appropriate considering the significant time that would be required to reach the WF3 specified limit of 2% room volume concentration. An allowed outage time of 14 days for air handling AH-5 with one or both Administrative Area air mixers non-functional is appropriate considering the additional 16 days available until the WF3 specified limit of 2% room volume concentration was reached.

[4] An outage of one air mixer is conservatively treated with the same urgency as an outage of both air mixers.

	In order to ensure monitoring of AH-5 and air mixer operation post-DBA, AR 00229188 is issued to update the appropriate procedures to add steps to monitor AH-5 and air mixer operation and take corrective actions as necessary consistent with the time limits provided by calculation 5L2, and summarized in the Topic Notes. Industry guidance regarding manual operator actions is provided by Information Notice 97-78, Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994. According to this guidance, a new operator action during a DBA would be subject to enhanced scrutiny due to the high consequence of operator error and the assumed operator action times which are inputs to the safety analyses. However, this heightened sensitivity to what the operators are doing is required only during the first few hours of a DBA when minimal staffing is assumed and there are many time-critical operator actions to perform. The action required to compensate for the absence of AH-5 operation is one which can be performed as much as a week after the initial DBA. In addition, it is an action which could be performed by one of the many additional personnel (not necessarily an operator) who would be on site to assist in post-DBA recovery efforts. Therefore, the proposed changes do not create the possibility of an accident of a different type than 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: EC 47846 will install Valve-Regulated Lead–Acid (VRLA) batteries in the TSC. These batteries recombine hydrogen gas, but can emit small amounts of hydrogen gas. Calculation 5-F, Control Room / Computer Area HVAC Cooling Loads determines the maximum hydrogen generation rate in the TSC during normal operations (float charge). Under ideal conditions (no ventilation and air-tight room), it would take approximately 276 days to reach the hydrogen accumulation limit of 2% of the room air volume specified by IEEE 1187-2002 for hydrogen accumulation. However, the TSC is continually ventilated during all normal and design basis accident conditions. The only exception is the control room isolation mode of operation. For this condition, the batteries during their entire lifecycle cannot produce enough hydrogen to reach 2% because the ventilation system recirculates the control room envelope air, allowing the entire volume of the CRE to dilute the hydrogen. EC 47846 will also install VRLA batteries in the RAB +7' Elevation Work Area. These batteries recombine hydrogen gas, but can also emit small amounts of hydrogen gas. Calculation 5L2, Office Area – HVAC Cooling Loads determines the maximum hydrogen generation rate in the RAB +7' Elevation Work Area during normal operations (float charge). This area is ventilated and there will be not be any hydrogen accumulation during normal plant conditions. A LOOP would shut down normal ventilation which is not powered by redundant, safety-related sources. However, the LOOP would not stop the VLRA batteries from potentially creating hydrogen gas. Under these conditions (no ventilation with batteries emitting hydrogen), it would take approximately 242 days to reach the hydrogen accumulation limit of 2% of the room air volume. In addition, a fan powered by the same circuit as the UPS which is charging the batteries will be circulating and mixing air in the area. Based on these conditions it is concluded that H2 will not accumulate in any measureable amount. The Work Area contains Safety Related power cables which could possibly fail in the event of a hydrogen gas explosion – leading to 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. However, the evaluation provided by calculation 5L2 in conjunction with procedure changes to ensure that the Work Area is adequately ventilated (to remove or dilute hydrogen) for any condition which would have AH-5 shut down for any appreciable length of time demonstrates that circumstances that lead up to an explosion will not occur. The addition of the design function for the RAB Air Conditioning System and the Control Room Air Conditioning System to limit the accumulation of hydrogen to a level below 2% does not result in any altered operating procedure or system configuration for the affected systems or any other system. Therefore, the installation of batteries in RAB +7 Elevation Work Area and the TSC will not create the possibility for a malfunction of an SSC important to safety with a different result than any previously		

	evaluated in 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 does not have any interaction with the Fuel Cladding, Reactor Coolant System Pressure Boundary, or the Containment Vessel. The proposed activities for installation of VLRA batteries in RAB +7' Elevation Work Area and TSC does not affect any parameters related to design basis limit for a fission product barriers, as no dose calculations or analysis is being altered with the proposed change 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 proposed activity is a physical change and does not involve a change to an element of a UFSAR described evaluation methodology, nor does it use an alternate evaluation methodology used in establishing the design basis or used in the safety analysis. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR that is used to establish the design bases or used 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. #:** 15-03 / 0

Proposed Change / Document: EC58901 (Evaluate RC IT0125-1 for Use as CPC Input)

Description of Change: CR-WF3-2015-04548 identified a Core Protection Calculator (CPC) trip on Channel A. Subsequent trouble shooting identified Reactor Coolant System (RCS) cold leg temperature loop RC IT0122CA (REACTOR COOLANT LOOP 2A COLD LEG TEMP LOOP) as initiating the trip. EC58901 provides an equivalency for RC IT0125-1 (REACTOR COOLANT COLD LEG 2A TEMP LOOP) to provide Reactor Coolant System (RCS) cold leg 2A temperature input into the core protection calculators (CPCs) channel A, Control Room Indicator RC ITI0102CA (CP-7), Control Room Recorder RC ITR0102CA (CP-7), and Plant Monitoring Computer (PMC) point A12118 as a substitute for input RC IT0122CA.

Summary of Evaluation:
EC58901 provides an equivalency for RC IT0125-1 as a substitute for input RC IT0122CA. CPC channel A was the primary impact evaluated as part of this change.

UFSAR Section 7.2.1.1.2.5 and W3-DBD-012 Section 2.2.1.2 describe the Core Protection Calculator System (CPC). The CPC system monitors process inputs to compute the Low Departure from Nuclear Boiling Ratio (DNBR) and Local Power Density (LPD). The calculated DNBR and LPD are compared with trip set points for initiation of a low DNBR trip and a high LPD trip. Four independent CPCs are provided, one in each protection channel (Channels A, B, C, and D). Calculation of DNBR and LPD is performed in each CPC. If a trip setpoint is exceeded, the channel sends a reactor trip signal to the Reactor Protective System (RPS). The RPS circuitry will shut down the reactor upon receiving trip signals from two of four CPC channels. The channels are physically and electrically isolated from each other and receive AC power from four redundant vital instrument buses.

EC58901 compared RC IT0125-1 and RC IT0122CA critical characteristics to validate equivalency between the two. The RC IT0122CA RTD assembly is a Weed (Ultra Electronics) model N9004E-2B. The RC IT0125-1 RTD assembly is a Weed model 9004D-2B. The critical characteristics evaluated were RTD Material, RTD Wiring Configuration, RTD Wiring Material, Specified RTD range, Specified RTD resistance, Temperature Coefficient, Specified Accuracy, Self-Heating Effect, Specified Response time, QDC Connector, and seismic. The cable to the RTD was also evaluated and found to be equivalent (safety related and EQ qualified). EC58901 determined that RC IT0125-1 will perform equivalently to RC IT0122CA with one exception. EC58901 identified that the cable route from RC ITE0125-1 to containment penetration 140 may not meet minimum separation requirements and the raceways may not be seismically qualified. EC58901 identified that this condition results in an operable but degraded/nonconforming condition in accordance with NRC IMC 0326 [Reference 4].

The 10CFR50.59 evaluation demonstrated that this change is acceptable. Each of the 8 questions specifically addressed the proposed change and associated impacts. While the 10CFR50.59 demonstrated this configuration is allowable, it is acknowledged that the operable but degraded/nonconforming condition should be corrected in a time frame commensurate with the safety significance of the condition.

[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. Waterford 3 Final Safety Analysis Report, Revision 308.
2. Waterford 3 Technical Specification through Amendment 242.
3. NEI 96-07 Revision 1, Guidelines for 10CFR50.59 Implementation.
4. NRC Inspection Manual IMC-326, OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS FOR CONDITIONS ADVERSE TO QUALITY OR SAFETY, 1/31/14.

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 Steelman / / SMI / Engineering / 7-24-15
Name (print) / Signature / Company / Department / Date

Reviewer: Tom Fleischer / / Entergy / Engineering / 7-24-15
Name (print) / Signature / Company / Department / Date

OSRC: Brian Lanka / / 7-24-15
Chairman's Name (print) / Signature / Date

15-12
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:

EC58901 provides an equivalency for RC IT0125-1 (REACTOR COOLANT COLD LEG 2A TEMP LOOP) to provide Reactor Coolant System (RCS) cold leg 2A temperature input into the core protection calculators (CPCs) channel A, Control Room Indicator RC ITI0102CA (CP-7), Control Room Recorder RC ITR0102CA (CP-7), and Plant Monitoring Computer (PMC) point A12118 as a substitute for input RC IT0122CA (REACTOR COOLANT LOOP 2A COLD LEG TEMP LOOP).

RCS cold leg 2A temperature indications on control panel indication (CP-7), control panel recorder (CP-7), and Plant Monitoring Computer (PMC) point A12118 provide operators information only. These indications do not automatically initiate any actions. These indications are not Technical Specification 3/4.3.3.5 (Remote Shutdown Instrumentation) or 3/4.3.3.6 (Accident Monitoring Instrumentation) required instruments. Redundant RCS cold leg temperature indications exists for use in operator decision making. Thus, the RCS cold leg 2A temperature indications have no impact on the accident frequency.

CPC channel A is required post-accident to initiate a reactor trip to limit event consequences. The CPCs are credited as a post-accident mitigation system. If CPC channel A were to trip prior to any event, no plant transient would occur because the CPC trip logic requires 2 channels out of 4 to initiate a reactor trip (2 channels out of 3 if one channel is bypassed). Thus, CPC channel A has no impact on the accident occurrence frequency.

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

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

☐ Yes
☒ No

BASIS:

The term "malfunction of an SSC important to safety" refers to the failure of structures, systems and components (SSCs) to perform their intended design functions.

UFSAR Section 7.2.2.1 states that the Reactor Protection System (RPS) which includes the CPC is designed to provide the following protective functions:

a) Initiate automatic protective action to assure that acceptable RCS and fuel design limits are not exceeded during specified anticipated operational occurrences.
b) Initiate automatic protective action during certain postulated accident conditions to aid the ESFS in limiting the consequences of the accident.

EC58901 determined that the change involves substitution of one type of component for another of similar function, with all applicable design and functional requirements continuing to be met and all new failure modes are bounded by the existing analysis with one exception. EC58901 identified that the cable route from RC ITE0125-1 to containment penetration 140 may not meet minimum separation requirements and the raceways may not be seismically qualified. EC58901 identified that this condition results in an operable but degraded/nonconforming condition.

Per Regulatory Guide 1.75, the purpose of maintaining minimum separation is to comply with the portion of GDC 21 stating "independence designed into protective systems be sufficient to ensure that no single failure results in loss of the protective function." The single failure in this case is one that can physically cause failure to two or more channels. Such a failure can be the result of a missile, or an event related jet, fire, or any other cause where the lack of physical separation can result in simultaneous failure of two or more channels.

Failure of the CPC RTD measurement channels will result in one of the following conditions:

- RTD measurement circuit opens: results in very high resistance and subsequent out of range high CPC trip.
- RTD measurement circuit shorts: results in very low resistance and subsequent out of range low CPC trip.
- RTD Compensating leg open circuit: results in no temperature compensation, RTD measurement circuit will read a step change low and will initiate a low out-of-range trip.
- RTD compensating leg short circuits with one of the measurement leg wires: results in equalizing the 3-wire bridge circuit resulting in a step change low and will initiate a low out-of-range trip.
- One or more leads short to ground: one of the effects identified above will occur, initiating a CPC trip.

All of the failure modes will initiate a CPC LPD, DNBR, VOPT, cold leg ΔT_{COLD} (ASGT), or temperature out-of-range trip. The same failure modes could occur if a non-seismic raceway failed as a result of a seismic event. No failure mode exists that would result in a continuation of an accurate temperature input under actual design basis event conditions and prevent a reactor trip when required.

Since, all failure modes result in the CPC channel A continuing to meet the safety function of tripping the channel which is the required CPC protective action, this change does not result in more than a minimal increase in likelihood of occurrence of malfunction of an SSC important to safety.

A review of past Waterford 3 operating experience (condition reports and work orders) identified only one past calibration failure which was due to a human performance error. The RC IT0125-1 loop has been reliable which means it is not likely to fail in this configuration.

RCS cold leg 2A temperature indications on control panel indication (CP-7), control panel recorder (CP-7), and Plant Monitoring Computer (PMC) point A12118 provide operators information only. These indications do not automatically initiate any actions. These indications are not Technical Specification 3/4.3.3.5 (Remote Shutdown Instrumentation) or 3/4.3.3.6 (Accident Monitoring Instrumentation) required instruments. Redundant RCS cold leg temperature indications exists for use in operator decision making. Thus, the RCS cold leg 2A temperature indications are not important to safety.

Thus, the proposed change does not result in more than a minimal 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:

When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators. The table below shows all the UFSAR Chapter 15 accidents which credit CPC trips. The CPC trips and auxiliary trips using RCS cold leg temperature input are DNBR, LPD, VOPT, Cold Leg ΔT, and cold leg temperature out of range.

Event	Chapter 15	CPC Trip	Resulting Containment Environment
Decrease in feedwater temperature	15.1.1.1	DNBR	Mild
Increase in feedwater flow	15.1.1.2	DNBR	Mild
Increase in Mainsteam flow	15.1.1.3	VOPT, DNBR, LPD	Mild
Increased MS flow with a concurrent LOOP	15.1.2.3	VOPT, DNBR, LPD	Mild
Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component	15.1.2.4	CPC RCP Speed	Mild
Steam line break- post trip return to power	15.1.3.1	VOPT	Harsh Response in 5.0 seconds (bounding case)
Steam line break- pre-trip power excursion	15.1.3.3	DNBR, VOPT	Harsh Response in 4.53 seconds
Loss of Normal AC Power	15.2.1.4	DNBR	Mild
Feedwater Line Break Large Break Small Break	15.2.3.1 15.2.3.1.3.1.2 15.2.3.1.4.3.2	DNBR No DNBR credited No DNBR credited	Harsh *CPC trip not credited
Partial Loss of Forced Reactor Coolant Flow	15.3.1.1	CPC RCP Speed	Mild
Total Loss of Forced Reactor Coolant Flow	15.3.2.1	CPC RCP Speed	Mild
Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	15.3.3.1	CPC RCP Speed	Mild
Uncontrolled CEA Withdrawal from Subcritical Conditions	15.4.1.1	CPC Bypass Removal	Mild
Uncontrolled CEA withdrawal at HZP	15.4.1.2	VOPT	Mild

CEA Withdrawal at Power	15.4.1.3	DNBR, VOPT	Mild
Control Element Assembly Misoperation	15.4.1.4	DNBR	Mild
Uncontrolled CEA Withdrawal form a Subcritical Condition	15.4.1.7	CPC Bypass Removal	Mild
CEA Ejection at Full Power	15.4.3.2	VOPT	Harsh Response in 0.42 second
Primary Sample or Instrument Line Break	15.6.3.1	CPC Low Pressurizer Pressure Out of Range	Mild
Steam Generator Tube Rupture	15.6.3.2	CPC Hot Leg Saturation Trip	Mild
Asymmetric Steam Generator	15.9.1.1	Aux Trip: Cold leg ΔT	Mild

EC58901 evaluated each UFSAR Chapter 15 event and determined that RC IT0125-1 loop would be equivalent with RC IT0122CA. This means that the CPC credited trips will be unaffected and associated radiological consequences not be impacted. For events caused by seismic, jet impingement, or fire that could impact the RC IT0125-1 cable routing inside containment, the CPC safety function would be a tripped condition and these events would not prevent or delay the CPC from performing this intended safety function.

The RCS cold leg 2A temperature indications do not automatically initiate any actions. These indications are not Technical Specification 3/4.3.3.5 (Remote Shutdown Instrumentation) or 3/4.3.3.6 (Accident Monitoring Instrumentation) required instruments. Redundant RCS cold leg temperature indications exists for use in operator decision making. Thus, the RCS cold leg 2A temperature indications have no impact on the accident consequences.

The accidents previously evaluated in the UFSAR were assessed for potential impacts and no adverse consequences were identified. The proposed change will ensure the CPC is capable of performing its specified function. Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

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

BASIS:

When determining which malfunctions represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose.

EC58901 has the potential to impact the core protection calculators (CPCs) channel A, Control Room Indicator RC ITI0102CA (CP-7), Control Room Recorder RC ITR0102CA (CP-7), and Plant Monitoring Computer (PMC) point A12118.

EC58901 identified that the cable route from RC ITE0125-1 to containment penetration 140 may not meet minimum separation requirements and the raceways may not be seismically qualified. EC58901 identified that this condition results in an operable but degraded/nonconforming condition. Failure of the CPC RTD measurement channels will result in one of the following conditions:

- RTD measurement circuit opens: results in very high resistance and subsequent out of range high CPC trip.
- RTD measurement circuit shorts: results in very low resistance and subsequent out of range low CPC trip.
- RTD Compensating leg open circuit: results in no temperature compensation, RTD measurement circuit will read a step change low and will initiate a low out-of-range trip.
- RTD compensating leg short circuits with one of the measurement leg wires: results in equalizing the 3-wire bridge circuit resulting in a step change low and will initiate a low out-of-range trip.
- One or more leads short to ground: one of the effects identified above will occur, initiating a CPC trip.

UFSAR Table 7.2-5 provides the failure modes and effects analysis. UFSAR Table 7.2-5 provides the following applicable failure modes:

Component	Failure	Sheet
Cold Leg Temperature	Spurious Low	2
Cold Leg Temperature	Spurious High	3
Core Protection Calculator	Tripped	5
Core Protection Calculator	Failure to Trip	5

The EC58901 potential failure modes are consistent with those listed in UFSAR Table 7.2-5.

Technical Specification Section 2 Bases states that the DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.
a. RCS Cold Leg Temperature-Low > 495°F
b. RCS Cold Leg Temperature-High < 580°F

This means that the cold leg temperature loop failure modes would most likely result in a CPC cold leg temperature out of range condition and an associated CPC trip condition. The CPC trip is the required specified safety function, thus this change does not does 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.

The RCS cold leg 2A temperature indications do not automatically initiate any actions. These indications are not Technical Specification 3/4.3.3.5 (Remote Shutdown Instrumentation) or 3/4.3.3.6 (Accident Monitoring Instrumentation) required instruments. Redundant RCS cold leg temperature indications exists for use in operator decision making. Thus, the RCS cold leg 2A temperature indications have no impact on the malfunction consequences.

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

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

BASIS:

UFSAR Section 7.2.1.1.2.5 and W3-DBD-012 Section 2.2.1.2 describe the Core Protection Calculator System (CPC). The CPC system monitors process inputs to compute the Low Departure from Nuclear Boiling Ratio (DNBR) and Local Power Density (LPD). The calculated DNBR and LPD are compared with trip set points for initiation of a low DNBR trip and a high LPD trip. Four independent CPCs are provided, one in each protection channel (Channels A, B, C, and D). Calculation of DNBR and LPD is performed in each CPC. If a trip setpoint is exceeded, the channel sends a reactor trip signal to the Reactor Protective System (RPS). The RPS circuitry will shut down the reactor upon receiving trip signals from two of four CPC channels. The channels are physically and electrically isolated from each other and receive AC power from four redundant vital instrument buses.

EC58901 impacts core protection calculators (CPCs) channel A. A trip of CPC channel A would not adversely impact the plant because CPC trip logic requires 2 channels out of 4 to initiate a reactor trip (2 channels out of 3 if one channel is bypassed). If multiple CPC channels spuriously tripped, that would cause a reactor trip and would remain bounded by the UFSAR Section 15.2.1.4 (Loss of Normal AC). If CPC channel A failed to trip when required, the CPC contain sufficient redundancy such that the remaining channels would initiate a reactor trip. If the CPCs and the Reactor Protection System (RPS) all failed to initiate a reactor trip when required, this scenario would remain bounded by the UFSAR Section 15.8 Anticipated Transient Without Scram scenario.

The RCS cold leg 2A temperature indications do not automatically initiate any actions. These indications are not Technical Specification 3/4.3.3.5 (Remote Shutdown Instrumentation) or 3/4.3.3.6 (Accident Monitoring Instrumentation) required instruments. Redundant RCS cold leg temperature indications exists for use in operator decision making. Thus, the RCS cold leg 2A temperature indications cannot create an accident of a different type.

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

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

BASIS:
EC58901 compared RC IT0125-1 and RC IT0122CA critical characteristics to validate equivalency between the two. The RC IT0122CA RTD assembly is a Weed (Ultra Electronics) model N9004E-2B. The RC IT0125-1 RTD assembly is a Weed model 9004D-2B. The critical characteristics evaluated were RTD Material, RTD Wiring Configuration, RTD Wiring Material, Specified RTD range, Specified RTD resistance, Temperature Coefficient, Specified Accuracy, Self-Heating Effect, Specified Response time, and QDC Connector. The cable to the RTD was also evaluated and found to be equivalent (safety related and EQ qualified). EC58901 determined that RC IT0125-1 will perform equivalently to RC IT0122CA and thus the CPC safety function will not be adversely impacted.

EC58901 identified that the cable route from RC ITE0125-1 to containment penetration 140 may

not meet minimum separation requirements and the raceways may not be seismically qualified. EC58901 identified that this condition results in an operable but degraded/nonconforming condition. Failure of the CPC RTD measurement channels will result in one of the following conditions:

- RTD measurement circuit opens: results in very high resistance and subsequent out of range high CPC trip.
- RTD measurement circuit shorts: results in very low resistance and subsequent out of range low CPC trip.
- RTD Compensating leg open circuit: results in no temperature compensation, RTD measurement circuit will read a step change low and will initiate a low out-of-range trip.
- RTD compensating leg short circuits with one of the measurement leg wires: results in equalizing the 3-wire bridge circuit resulting in a step change low and will initiate a low out-of-range trip.
- One or more leads short to ground: one of the effects identified above will occur, initiating a CPC trip.

UFSAR Table 7.2-5 provides the failure modes and effects analysis. UFSAR Table 7.2-5 provides the following applicable failure modes:

Component	Failure	Sheet
Cold Leg Temperature	Spurious Low	2
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Core Protection Calculator	Failure to Trip	5

The EC58901 potential failure modes are consistent with those listed in UFSAR Table 7.2-5.

Technical Specification Section 2 Bases states that the DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.
a. RCS Cold Leg Temperature-Low > 495°F
b. RCS Cold Leg Temperature-High < 580°F

This means that the cold leg temperature loop failure modes would most likely result in a CPC cold leg temperature out of range condition and an associated CPC trip condition. The CPC trip is the required specified safety function, thus this 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.

The RCS cold leg 2A temperature indications do not automatically initiate any actions. These indications are not Technical Specification 3/4.3.3.5 (Remote Shutdown Instrumentation) or 3/4.3.3.6 (Accident Monitoring Instrumentation) required instruments. Redundant RCS cold leg temperature indications exists for use in operator decision making. The PMC point A12118 is used in procedures NE-004-004 (RCS Flow Rate Calculation with COLSS Inoperable) and NE-004-006 (RCS Flow Rate Calculation with COLSS Operable) for calculating the RCS flow value. This surveillance is performed during normal plant conditions in which EC58901 has demonstrated all critical characteristics are equivalent, thus the PMC point A12118 value and use will be consistent with the tolerance and uncertainty requirements for the indication. Thus, the RCS cold leg 2A temperature indications cannot create a malfunction with different results.

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 potentially impacted fission product barriers are associated with the specified acceptable fuel design limits (DNBR and Linear Heat Rate). The CPC specified safety function is to prevent anticipated operational occurrences from exceeding a fuel design limit and to limit the consequences of accident scenarios. EC58901 equivalency determined that the CPC channel A will still perform its specified function, thus this change has no impact on the fission product barrier design basis limits.

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:

Methods of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC [Reference 6 Section 3.10]. This change is not impacting the methodology or topical reports used in the UFSAR analyses. The existing UFSAR evaluations utilizing the CPC as a mitigating system have 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.

I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # / Rev. #:** 2015-04 / 0

Proposed Change / Document: EC54054 Cycle 21 Reload

Description of Change: EC54054 Cycle 21 Reload Process Applicability Determination (PAD) identified two adverse changes. The Cycle 21 reload analysis report had a physics assessment checklist (PAC) exception for the loss of coolant accident (LOCA) unrodded pin census and a new CEA Ejection analysis of record (AOR) was generated due to finding an error in the calculation process. This evaluation will address the two adverse changes.

Summary of Evaluation:

EC54054 Cycle 21 Reload Process Applicability Determination (PAD) identified two adverse changes. The first adverse change is associated with the Cycle 21 reload analysis report which had a physics assessment checklist (PAC) exception for the loss of coolant accident (LOCA) unrodded pin census. The core pin census exception is applicable to large break LOCA (LBLOCA) and the method of dispositioning the exception is provided as follows:

Core Pin Census – A bounding core pin census was used in the bounding LBLOCA core-wide oxidation analysis performed for next generation fuel (NGF). This pin census was designed to produce a limiting core-wide oxidation result for the AOR. However, this pin census did not bound the Cycle 21 specific core pin census data over the full range of fuel rod peaking factors. Therefore, the limiting core-wide oxidation case from the AOR was evaluated using the Cycle 21 specific core pin census data. The comparison of results showed that the bounding LBLOCA AOR core wide oxidation result remained bounding compared to the Cycle 21 specific value. The change in core pin census does not impact the maximum local cladding oxidation (MCO). The AOR MCO has been confirmed to remain bounding.

This means that the Cycle 21 specific core-wide oxidation remained below the LBLOCA bounding analysis and also less than the 10CFR50.46 acceptance criteria of 1%.

The second adverse change is that Westinghouse identified during the Control Element Assembly (CEA) ejection reanalysis for the CEA drop time project that the analysis of record (CN-TAS-06-41 Rev. 0) missed a step in the analysis procedure that led to non-conservative results when calculating the event fuel failure. The CEA ejection analysis of record did not appropriately include the Next Generation Fuel (NGF) critical heat flux correlation in calculating the event thermal margin and associated DNBR values. The new AOR (CN-SCC-15-004 Rev. 0) showed for Cycle 20 that the analysis fuel failure increased from 8.78% to 11.51% for the limiting fuel failure case. For Cycle 21, the CEA ejection fuel failure is 11.83% using the new AOR. The fuel failure limit used in the calculation for radiological doses for the CEA ejection event is 15.00%. Therefore, the event still meets radiological dose input limit with the new DNBR values and the radiological dose analyses remain valid.

Both adverse changes were demonstrated to be within the design requirements and do not adversely impact any information contained in the UFSAR.

[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. Waterford 3 Updated Final Safety Analysis Report, Through Amendment 308.
2. Waterford 3 Technical Specification, Through Amendment 245.
3. NEI 96-07 Revision 1, Guidelines for 10CFR50.59 Implementation, November 2000.

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 Steelman / EC54054 Signature / SMI / Engineering / 10-26-15
Name (print) / Signature / Company / Department / Date

Reviewer: Ralph Griffith / EC54054 / Entergy / Nuc Eng / 10-29-15
Name (print) / Signature / Company / Department / Date

OSRC: Brian Lanka / EC54054 / 11-12-15
Chairman's Name (print) / Signature / Date

W3 15-21
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:

EC54054 Cycle 21 Reload Process Applicability Determination (PAD) identified two adverse changes. The Cycle 21 reload analysis report had a physics assessment checklist (PAC) exception for the loss of coolant accident (LOCA) unrodded pin census and a new CEA Ejection analysis of record (AOR) was generated due to finding an error in the calculation process.

The LOCA unrodded pin census relates to UFSAR Section 15.6.3.3 (Loss of Coolant Accident) analysis. The LOCA unrodded pin census does not impact the occurrence of the accident but is relevant to the accident results and severity.

The CEA Ejection calculation issue relates to UFSAR Section 15.4.3.2 (CEA Ejection Accident). The CEA Ejection calculation issue does not impact the occurrence of the accident but is relevant to the accident results and severity.

The UFSAR Section 15.4.3.2 and 15.6.3.3 events assume the accident occurs. The reload analysis evaluates the impact of reload related parameters on the severity of the accident to ensure the results remain within predetermined limits. The adverse changes identified in the EC54054 PAD have no impact on the frequency of occurrence only the consequences.

In addition, no changes to plant equipment are required due to the Cycle 21 fuel design. The proposed change does not create any new system interactions that could cause an accident. Therefore, the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

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

☐ Yes
☒ No

BASIS:

The Waterford 3 Cycle 21 Batch EE reload fuel assemblies are acceptable for use in Waterford 3 Cycle 21. There is acceptable mechanical design margin for the Cycle 21 core containing Batch EE fuel assemblies and other resident fuel batches. The Batch EE fuel assemblies meet all required design and functional requirements. The nuclear design of Batch EE fuel was accomplished using NRC approved analysis methodologies under approved quality assurance programs. The performance of the Batch EE assemblies is not expected to be significantly different than previous fuel batches. The probability of fuel failure due to mechanical or flow induced vibration and fretting with the spacer grids [FSAR 4.2.1.2.1.g, 4.2.3.1.1, 4.2.3.1.3, 4.2.3.2.1 and 4.2.3.2.4] will not be increased.

The dimensions and placement of the guide tubes in the Batch EE fuel assemblies are the same as Batch DD fuel assemblies, and as such, there are no compatibility issues with CEAs.

The Cycle 21 fuel and core designs will not degrade the performance of any safety system assumed to function in the safety analyses, nor will these changes decrease the reliability of safety systems. Instrumentation accuracy or response characteristics will not be impacted.

All equipment important to safety will function in the same manner with the Cycle 21 reload core as with the previous core. There is no characteristic of the Cycle 21 core, with the Batch EE reload assemblies that would increase the probability of a malfunction of equipment important to safety. Therefore, the likelihood of occurrence of a malfunction of an SSC important to safety is not increased due to Cycle 21 core reload.

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

BASIS:

EC54054 Cycle 21 Reload Process Applicability Determination (PAD) identified two adverse changes. The first adverse change is associated with the Cycle 21 reload analysis report which had a physics assessment checklist (PAC) exception for the loss of coolant accident (LOCA) unrodded pin census. The core pin census exception is applicable to large break LOCA (LBLOCA) and the method of dispositioning the exception is provided as follows:

Core Pin Census – A bounding core pin census was used in the bounding LBLOCA core-wide oxidation analysis performed for next generation fuel (NGF). This pin census was designed to produce a limiting core-wide oxidation result for the AOR. However, this pin census did not bound the Cycle 21 specific core pin census data over the full range of fuel rod peaking factors. Therefore, the limiting core-wide oxidation case from the AOR was evaluated using the Cycle 21 specific core pin census data. The comparison of results showed that the bounding LBLOCA AOR core wide oxidation result remained bounding compared to the Cycle 21 specific value. The change in core pin census does not impact the maximum local cladding oxidation (MCO). The AOR MCO has been confirmed to remain bounding.

This means that the Cycle 21 specific core-wide oxidation remained below the LBLOCA bounding analysis and also less than the 10CFR50.46 acceptance criteria of 1%.

The second adverse change is that Westinghouse identified during the Control Element Assembly (CEA) ejection reanalysis for the CEA drop time project that the analysis of record (CN-TAS-06-41 Rev. 0) missed a step in the analysis procedure that led to non-conservative results when calculating the event fuel failure. The CEA ejection analysis of record did not appropriately include the Next Generation Fuel (NGF) critical heat flux correlation in calculating the event thermal margin and associated DNBR values. The new AOR (CN-SCC-15-004 Rev. 0) showed for Cycle 20 that the analysis fuel failure increased from 8.78% to 11.51% for the limiting fuel failure case. For Cycle 21, the CEA ejection fuel failure is 11.83% using the new AOR. The fuel failure limit used in the calculation for radiological doses for the CEA ejection event is 15.00%. Therefore, the event still meets radiological dose input limit with the new DNBR values and the radiological dose analyses remain valid.

Both adverse changes were demonstrated to be within the design requirements and do not adversely impact any information contained in the UFSAR.

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 Waterford 3 Cycle 21 reload safety analyses were performed to assure that acceptance criteria are met for fuel performance, thermal-hydraulic performance, LOCA ECCS performance and non-LOCA transient response. These analyses confirm that the Cycle 21 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 21 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 21. 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 21 reload.

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

BASIS:

The Batch EE fuel assemblies of the Cycle 21 core meet all required design and functional requirements. The nuclear design of Batch EE fuel was accomplished using NRC approved analysis methodologies under approved quality assurance programs. The performance of the Batch EE assemblies is not expected to be significantly different than previous fuel batches. The Cycle 21 reload core will not result in changes to the radiological release rate/duration, will not create new release mechanisms, and will not impact radiation release barriers. There are no new system interactions or connections associated with the Cycle 21 core reload.

There were no changes in the failure modes of equipment important to safety assumed in the design and analyses associated with the Cycle 21 reload. No initiators of any of the accidents already postulated are impacted by the Cycle 21 reload. Therefore, operation of Waterford 3 with the Cycle 21 core will not cause 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:
Installation of the Cycle 21 core will not cause the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR. Equipment important to safety will function in the same manner with the Cycle 21 core as with the Cycle 20 core. The impact of changes in core characteristics on any parameter that would affect the function of equipment important to safety has been accounted for in the analyses applicable for Cycle 21.

The Waterford 3 testing and verification program ensures that all required calibrations and setpoint changes resulting from the Cycle 21 reload design are performed. There are no new modes of failure associated with any of the changes for Cycle 21. No changes in the failure modes of the equipment important to safety were assumed in the Cycle 21 core design or fuel mechanical analyses. No changes due to the Cycle 21 reload analysis will significantly alter the

way in which Waterford 3 operates.

Based on the above, the possibility of a malfunction of equipment important to safety having a different result than any previously evaluated will not be created due to the fuel management, reload fuel assembly design changes, or other reload-related changes necessary to operate Cycle 21.

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 Waterford 3 Cycle 21 reload safety analyses were performed to assure that acceptance criteria are met for fuel performance, thermal-hydraulic performance, LOCA ECCS performance and non-LOCA response. These analyses confirm that the core can be operated safely and can be expected to meet license requirements for accident response. The Cycle 21 reload safety analyses were performed to demonstrate compliance with the existing design basis limits for the fuel cladding, Reactor Coolant System (RCS) pressure boundary, and containment.

The first adverse change is associated with the Cycle 21 reload analysis report which had a physics assessment checklist (PAC) exception for the loss of coolant accident (LOCA) unrodded pin census. The core pin census exception is applicable to large break LOCA (LBLOCA) and the limiting core-wide oxidation case from the AOR was evaluated using the Cycle 21 specific core pin census data. The comparison of results showed that the bounding LBLOCA AOR core wide oxidation result remained bounding compared to the Cycle 21 specific value. The change in core pin census does not impact the maximum local cladding oxidation (MCO). The AOR MCO has been confirmed to remain bounding.

The Cycle 21 specific core-wide oxidation remained below the LBLOCA bounding analysis and also less than the 10CFR50.46 acceptance criteria of 1%. This means that the fission product boundary was not adversely impacted with respect to the design limits.

The second adverse change is that Westinghouse identified during the Control Element Assembly (CEA) ejection reanalysis for the CEA drop time project that the analysis of record (CN-TAS-06-41 Rev. 0) missed a step in the analysis procedure that led to non-conservative results when calculating the event fuel failure. For Cycle 21, the CEA ejection fuel failure is 11.83% using the new AOR. The fuel failure limit used in the calculation for radiological doses for the CEA ejection event is 15.00%. Therefore, the event still meets radiological dose input limit with the new DNBR values and the radiological dose analyses remain valid with no adverse change to the fission product barrier.

All events have been evaluated in the reload analysis to assure that they meet their respective criterion for Cycle 21. 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 21.

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

BASIS:

Methods of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC [Reference 3 Section 3.10]. In accordance with Technical Specification 6.9.1.11.1, the Cycle 21 core was designed and evaluated using NRC approved analysis methodology under an approved quality assurance program. No new methodologies were required to verify that previous safety analyses are applicable to Cycle 21 or to perform reanalysis of any events.

The CEA ejection accident analysis issue was caused by an oversight in the implementation of the correct process to obtain the desired results. The new analysis did not change any methods previously approved by the NRC. Therefore, there has been no deviation from the methods of evaluation described in the UFSAR.

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

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

I. OVERVIEW / SIGNATURES[1]

Facility: WF3 **Evaluation # 2015-05 / Rev. #:** 0

Proposed Change / Document: Engineering Change EC61746, "Temporary Modification for Control Element Drive Mechanism Control System Position Switches"

Description of Change: This activity justifies the use of temporary modification 61746 from plant start up to and during full power. EC 61746 provides a simulated Control Element Assembly (CEA) 66 Reed Switch Position Transmitter (RSPT) 2 position indication to Control Element Assembly Calculator (CEAC) 2 in place of the actual position indication supplied by CEA 66 RSPT 2. This activity impacts the CEAC 2 input only and does not affect the Core Protection Calculator (CPC) monitoring of target rod indications. CEA 66 is part of regulating group 3 and subgroup 15. Regulating group 3 is normally fully withdrawn during power operation. CEA 66 RPST 2 indication will be considered inoperable with the temporary modification installed.

Summary of Evaluation: This activity justifies the use of a temporary modification installed under EC 61746 R0 during power operations and during plant startup. All eight questions below have been answered "no" based on the justifications provided. Prior Commission approval of the activity as described above is therefore not required.

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.

Not Applicable.

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

Preparer: Thomas Fleischer / See EC61746 / EOI / DE I&C / 12-2-2015
Name (print) / Signature / Company / Department / Date

Reviewer: Joel Rachal /See EC 61746 / EOI/ Training / 12-3-2015
Name (print) / Signature / Company / Department / Date

OSRC: / Brian Lanka / [signature] / 12/4/15
Chairman's Name (print) / Signature / Date

OSRC Meeting # 15-23

[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.

II. 50.59 EVALUATION

This activity justifies the use of the temporary modification installed under EC 61746 R0 during power operations and during plant startup.

The safety function of the RSPTs is to provide control rod position signals to the Core Protection Calculators (CPCs) and the CEA Calculators (CEACs) for the calculation of DNBR and LPD trip penalties.

As described in FSAR 7.2.1.1.2.5, the CPCs use the CEA group position information to calculate penalty factors for CEA subgroup deviations and CEA groups out of sequence. This indication is based on selected "target" CEA positions assigned to each CPC channel.

CEA 66 RSPT 2 indication is a non-target CEA indication. "Non-target" CEA RSPT and *"target" CEA RSPT position signals input to CEA calculators (CEACs). The CEACs calculate* penalty factors for deviation of one or more individual CEA positions from a subgroup and transmit the appropriate deviation penalty factor to the CPC. The penalty factors calculated by the CEACs and/or CPCs cause CPC DNBR and LPD trip margins to be reduced, thereby assuring conservative operation of the RPS.

CPC DNBR and LPD penalty factors are derived based on the following detected conditions:
1) Single CEA deviation in a subgroup calculated by CEA calculators
2) Subgroup deviations in a group calculated by CPCs
3) Groups out of sequence calculated by CPCs

From FSAR 7.2.2.1.1, CEA design requirements were based on the following anticipated operational occurrences (AOOs):
- Uncontrolled sequential withdrawal of CEA groups
- Out-of-sequence insertion or withdrawal of CEA groups
- Excessive sequential insertion of CEA groups
- Uncontrolled insertion or withdrawal of a CEA subgroup
- Dropping of one CEA subgroup
- Misalignment of CEA subgroup comprising a designated CEA group
- Uncontrolled insertion or withdrawal of a single CEA
- Dropped CEA
- A single CEA sticking, with the remainder of the CEAs in that group moving
- A statically misaligned CEA

Another function of the Core Element Assembly Calculators as described in FSAR Section 7.5.1.6 is to display the position of each regulating and shutdown CEA to the operator in a bar chart format on a dedicated CRT on the RTGB. This input is driven by the RSPT indications supplied to the Core Element Assembly Calculators. The operator has the capability to select either Core Element Assembly Calculator for display. In addition, a backup readout is provided that can be utilized to read the output of any CEA analog reed switch position signal. The backup readout is a digital meter on the CPCs operator's module, from which the operator can address any analog position signal for display on the digital meter. In addition to the displays, CEA deviation information is provided by the CEA calculators to the CPCs and a CEA deviation alarm. The CEA deviation alarm is provided to the plant annunciator system.

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

Does the proposed Change:

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

   BASIS:

   Accidents listed in the UFSAR associated with CEAs include CEA misoperation, CEA drop, CEA ejection, and uncontrolled CEA withdrawal (from power and low power conditions). The proposed temporary change to simulate an RSPT 2 signal for CEA 66 does not in any way impact the Control Element Drive Mechanism Control System (CEDMCS) ability to move or hold the CEA. The sequencing permissives for allowing CEA group movement are controlled by PMC programs using pulse counter inputs which are not affected by the RSPT indications. This proposed activity therefore does not increase the frequency of occurrence of an accident previously evaluated in the FSAR.

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 will result in a configuration that represents actual CEA 66 position when fully withdrawn, which occurs during normal operation

   A simulated signal input will be provided into CEAC-2 to simulate CEA-66 position. The simulation signal circuit consists of three resistors which mimic the actual resistor network on the RSPT. The existing power supply in the CPC/CEAC channel C cabinet is used for the voltage input. The circuit will be seismically mounted and located at the signal terminal block which will not interfere with any existing equipment. The resistors are ½ watt, therefore there is a negligible increase in heat generation within the cabinet. The resistor values are comparable to those on the RSPT circuit; therefore there will be no net increase in electrical load on the power supplies. A Parts Classification

Evaluation was performed on the RSPT substitution box which classifies it for use in a safety related system. The reliability of the resistors is comparable to that of the resistors in an RSPT.

If the simulated input signal fails or deviates from the normal range of CEAC inputs, CEAC 2 will continue to function as though an actual reed switch position transmitter had had failed. These failures are described in FSAR Table 7.2-5 and the effects remain bounded by the description currently in the FSAR.

During start-up, CEA 66 RSPT 2 will indicate CEA 66 is fully withdrawn. Other CEAs in this group will be indicated as fully in. This will result in a CEA position mismatch and the appropriate penalty factor will be calculated by CEAC 2 and applied by the CPCs as a trip margin reduction. This is a configuration similar to an uncontrolled CEA withdrawal from subcritical power as described in FSAR 15.4.1.1. If the mismatch remains in place after the plant exceeds 1E-4% power, the reactor will trip as described in FSAR 15.4.1.1 and 7.2.

The current licensing basis accommodates the failure of one Reed Switch Position Transmitter based on review of Technical Specification 3.1.3.2. Site procedure OP-901-102 provides guidance for operation with a failed input to the Core Element Assembly Calculators to ensure that the CEACs are capable of detecting an excessive outward deviation within a CEA subgroup with a failed CEA. Additional, sensor failures will be alarmed in the main control room and addressed in accordance with OP-901-102.

Based on the discussion above, this activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of the CEAC or CPC systems.

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

BASIS:

The only accident analysis at issue for the proposed temporary change is the CEA misoperation event. Core power distribution assessed by the CEA calculators (CEACS) and the subsequent augmented power distribution penalty factor is supplied as input to the core protection calculators (CPCs).

A CEA misoperation is defined as any event which could result from a single malfunction in the reactivity control system, with the exception of sequential group withdrawals which are considered in FSAR Subsections15.4.1.1 through 15.4.1.3. Group insertions (out of sequence) are adequately covered by CPC monitoring (CEA 66 RSPT 2 is not targeted to a CPC channel). Consequences of these events are not impacted by the proposed activity.

The single CEA drop is the most adverse CEA misoperation event discussed in FSAR

15.4.1.4 in terms of approach to the SAFDLs since dropped CEA subgroup is symmetric and produces less power distribution distortion. With the temporary modification installed an inadvertent insertion of CEA 66 while critical (including CEA drop) would not be sensed by Core Element Assembly Calculator 2.

No credit is taken for a reactor trip generated by the Core Protection Calculators in regards to a rod drop. FSAR section 15.4.1.4.2.1 states "Sufficient initial thermal margin is preserved by the Limiting Conditions for Operation (LCOS) so that the plant may experience a CEA drop without requiring a reactor trip to ensure that the SAFDLs are not violated."

In regards to a deviation within an individual subgroup, the Core Element Assembly Calculators are credited with detecting individual rod deviations and generating a penalty factor for excessive outward deviations. CEA-66 will be at the top of the core (approximately 150") during the period that the temporary modification is installed. With the CEA at this position, it is physically impossible for the CEA to have an outward deviation from its subgroup that would require generation of a penalty factor. If Regulating Group 3 containing CEA 66 is moved below 140", existing procedural guidance requires declaring CEAC 2 inoperable.

TS 3/4.1.3 Limiting Conditions for Operation already account for a single CEA failed sensor and therefore accommodate with sufficient margin an inadvertent operation or drop of a CEA monitored by a single CEAC channel. This assures the CEAC and CPC will continue to assess core power and apply the necessary penalties/trips as necessary to protect SAFDLs as credited in FSAR 15.4.1.4. This activity therefore will not have more than a minimal increase in the consequences of a misoperated or dropped CEA as described 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 consequences of CPC / CEAC or CEA malfunctions are no more severe with the simulated RSPT signal applied than with a normal live input. Failures to CEA position input signals are anticipated as listed in FSAR table 7.2-5, Failure Modes and Effect Analysis, which lists possible failure modes for non-target CEA position indications. The proposed temporary change results in no higher dose consequences due to a malfunction of a structure, system, or component than what has been previously evaluated.

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

BASIS:

The application of a simulated RSPT signal for a regulating group 3 CEA in place of a live signal does not introduce any new accident initiator or result in the CPC / CEAC system operating in a different fashion than would normally be the case. The CPC / CEAC system includes in its design, the capability to accommodate sensor failures (refer to FSAR Table 7.2-5). Failure of the simulated signal would be perceived by and acted upon in the same manner as the CPC / CEAC system currently treats actual failures of RSPT inputs. Failures to CEA position input signals are anticipated as listed in FSAR table 7.2-5, Failure Modes and Effect Analysis, which lists possible failure modes for non-target CEA position indications. The proposed activity has no potential for creating additional accidents of a similar frequency or magnitude as those currently contained in the licensing basis.

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 application of a simulated RSPT signal for a regulating group 3 CEA in place of a live signal does not introduce any new accident initiator or result in the CPC / CEAC system operating in a different fashion than would normally be the case. No new failure modes are created for the system. The CPC / CEAC system includes in its design, the capability to detect and accommodate sensor failures.

A failure of the simulated signal would be perceived by and acted upon in the same manner as the CPC / CEAC system currently treats actual failures of RSPT inputs. Failures to CEA position input signals are anticipated as listed in FSAR table 7.2-5, Failure Modes and Effect Analysis, which lists possible failure modes for non-target CEA position indications.

In the case where CEA 66 actual position may deviate from the simulated input signal, the simulated RSPT 2 sensor input would be perceived as being within range even though the input may not indicate actual rod position. This situation would not be unique to this rod and would be analogous to a CEA with an actual malfunctioning RPST that allows the sensor input to stay within range (stuck reed switch). This situation is described in FSAR table 7.2-5 Failure Modes and Effects Analysis as a target CEA with inaccurate positon. CEAC response would remain as described in the FSAR.

Therefore, the CEAC will continue to function as designed to detect and respond to sensor inputs that are not valid.

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 consequences of the CEA mis-operation events remain bounding with application of the proposed temporary change and the CPC / CEAC system response to other accidents remains unchanged. As such, the CPC/CEAC system continues to satisfy all requirements for protecting against design basis events. This provides assurance that the design basis limits for fission product barriers described in the UFSAR will not be exceeded. The proposed temporary change does not alter these limits in any way.

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 temporary change does not relate to an analytical methodology used to demonstrate compliance with required design bases.

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]

2016-01 KB 4/28/16

Facility: Waterford 3 Nuclear Generating Station **Evaluation # / Rev. #:** 0 / 0

Proposed Change / Document: LBDCR 16-0006 UFSAR Section 7.7.1.2.2 and 9.3.4 Change

Description of Change: CR-WF3-2015-6448 utilized a compensatory measure which required maintaining two charging pumps in the "ON" controller position. CR-WF3-2016-0243 identified that the CR-WF3-2015-6448 Process Applicability Determination (PAD) should have been a full 10CFR50.59 evaluation due to the conflict with UFSAR Section 9.3.4.1.2.1.2 stating that normally, one charging pump is operated, but two or three pump operations can be selected for higher purification flow if desired. The conflict is more than one charging pump was being operated and it was not required for higher purification but due to an equipment issue.

LBDCR 16-0006 will update UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) and Section 9.3.4 (Chemical and Volume Control System) to resolve the CR-WF3-2016-0243 identified conflict and to make USFAR Section 9.3.4 consistent with UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) which states automatic control is normally used during operation, but manual control may be used at any time.

Summary of Evaluation:
This evaluation demonstrated that the manual operation of the pressurizer level control system (charging pumps and letdown control valves) was already within the assumptions of the transient and accident analysis. This change is consistent with UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) which states automatic control is normally used during operation, but manual control may be used at any time. This change is within the current design and licensing basis and only updates the UFSAR information to make it consistent.

This change is not a new safety related operator action because the pressurizer level control system has no safety related credited function as specifically described in UFSAR Section 7.7 and NUREG-0787 Section 7.7. UFSAR Section 15.0.2 states the following:

> Systems which are not required to perform safety functions are described in Section 7.7. In general, normal automatic operation of these control systems is assumed unless lack of operation would make the consequences of the incidents more adverse. In this case, the particular control system is assumed to be inoperative. No credit is taken in the analyses for any operator action prior to initiation of the event which would mitigate the consequences of the transient; however, the analyses are performed on the basis that the plant is being operated within all limiting conditions for operation (LCO) at the initiation of all events.

As stated, the automatic operation of the pressurizer level control system is not credited to perform any safety related function. The transient and accident analyses do assume the most adverse control of the pressurizer level control system to maximize the event consequences.

The charging system is safety related and is required to perform a post-accident function. UFSAR Section 9.3.4.3 (Safety Evaluation) describes that the capability of the charging system to borate is not compromised by stopping letdown flow. Because safe shutdown can be achieved without letdown flow, letdown flow does not have a post-accident function. For accidents which involve a safety injection actuation signal or containment isolation actuation signal, the letdown line is automatically

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isolated which is a safety function. This change assumes operation within the restrictions contained in the Technical Specifications, Core Operating Limits Report, and Technical Requirements Manual.

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

References

1. Waterford 3 Updated Final Safety Analysis Report, Through Amendment 308.
2. Waterford 3 Technical Specification, Through Amendment 246.
3. Waterford 3 Technical Requirements Manual, Through Amendment 134.
4. NEI 96-07 Revision 1, Guidelines for 10CFR50.59 Implementation, November 2000.
5. NUREG-0787, Waterford 3 Safety Evaluation Report, July 1981.
6. NUREG-0787 Supplement 1, Waterford 3 Safety Evaluation Report, October 1981.
7. NUREG-0787 Supplement 8, Waterford 3 Safety Evaluation Report, December 1984.
8. W3F1-2001-0088, Amendment Request NPF-38-239 Revision of Letdown Line Break Dose Consequences, October 15, 2001.
9. W3F1-2002-0072, Supplement to Amendment Request NPF-38-239 Revision of Letdown Line Break Dose Consequences, August 27, 2002.
10. NRC Operating License Amendment 184, Waterford Steam Electric Station Unit 3 – Issuance of Amendment Re: Letdown Line Break Dose Consequences Revision, January 8, 2003.
11. W3F1-2004-0053, Amendment Request NPF-38-256 Alternate Source Term, June 15, 2004.
12. W3F1-2004-0071, Supplement to Amendment Request NPF-38-256 Alternate Source Term, August 19, 2004.
13. W3F1-2004-0101, Supplement 4 to Amendment Request NPF-38-256 Alternate Source Term, October 19, 2004.
14. NRC Operating License Amendment 198, Alternate Source Term, March 29, 2005.

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 Steelman / [signature] / SMI / Engineering / 3-30-16
Name (print) / Signature / Company / Department / Date

Reviewer: John Taylor / [signature] / Entergy / Engineering / 3-30-16
Name (print) / Signature / Company / Department / Date

OSRC: / Brian Lanka / [signature] / 4-1-16
Chairman's Name (print) / Signature / Date

OSRC Meeting # 16-03

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 addresses updating UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) and Section 9.3.4 (Chemical and Volume Control System) to resolve the CR-WF3-2016-0243 identified conflict and to make USFAR Section 9.3.4 consistent with UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) which states automatic control is normally used during operation, but manual control may be used at any time. This change will allow manual control of the pressurizer level control system for more than just increased purification flow. The normal method of controlling pressurizer level in automatic is still preferred and is not changed by this update.

The UFSAR was reviewed to identify which accidents previously evaluated could be initiated or caused by the proposed change. The pressurizer level control system is evaluated in the UFSAR as an accident initiator. The accident analyses that are already included in the UFSAR are as follows:

UFSAR Appendix 5.2B (Low Temperature Overpressure Protection During Heatup and Cold Shutdown) demonstrates that the shutdown cooling system overpressure protection meets the low temperature overpressure protection criteria by analyzing the limiting transients.

UFSAR Section 15.4.1.5 (CVCS Malfunction - Inadvertent Boron Dilution) is a malfunction which results in unborated water being pumped at the maximum possible rate into the RCS by the demineralized water supply system is assumed to occur. For this to occur, one or more charging pumps must be on, the primary water makeup water pumps must be on, and the demineralized water supply system must be aligned to supply water to the charging pump suction via the volume control tank. Since at least three simultaneous equipment malfunctions would be required to produce the above conditions, the incident could only be the result of improper operator action accompanied by a single equipment malfunction.

UFSAR Section 15.5.1.1 (Chemical and Volume Control System Malfunction) addresses an accident caused by the startup of all backup charging pumps with letdown flow at minimum.

UFSAR Section 15.5.2.1 (Chemical and Volume Control System Malfunction with a Concurrent Single Failure of an Active Component) addresses an accident caused by the startup of all backup charging pumps with letdown flow isolated.

UFSAR Section 15.4.1.5 (CVCS Malfunction - Inadvertent Boron Dilution) is classified as a moderate frequency incident [Reference 1]. UFSAR Section 15.5.1.1 (Chemical and Volume Control System Malfunction) is classified as a moderate frequency incident. UFSAR Section 15.5.2.1 (Chemical and Volume Control System Malfunction with a Concurrent Single Failure of an Active Component) is classified as an infrequent incident. UFSAR Appendix 5.2B (Low

Temperature Overpressure Protection During Heatup and Cold Shutdown) does not list a frequency but would be consistent with the UFSAR Section 15.5.1.1 event.

UFSAR Section 15.0.1 defines a Moderate Frequency Incidents as incidents any one of which may occur during a calendar year for a particular plant. Infrequent Incidents are defined as incidents any one of which may occur during the lifetime of a particular plant.

These events already assume an operator error or equipment failure caused the chemical and volume control inventory inputs (charging) to be on and the chemical and volume control outputs (letdown) to be off or minimized. The equipment remains designed to the same NRC requirements, design, material, and construction standards as contained in the UFSAR. The use of manual control was already allowed by the UFSAR and all manual actions are procedurally controlled. The pressurizer level control system is already a system not required for safety. This change does not impact the initiating event assumptions contained in UFSAR Appendix 5.2B, Section 15.4.1.5, Section 15.5.1.1, and Section 15.5.2.1. This means the event frequency categories would not be impacted. Therefore, the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

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

BASIS:

The term "malfunction of an SSC important to safety" refers to the failure of structures, systems and components (SSCs) to perform their intended design functions.

This change impacts the automatic operation of the pressurizer level control system which has no safety related credited function as specifically described in UFSAR Section 7.7 and NUREG-0787 Section 7.7. UFSAR Section 15.0.2 states the following:

> Systems which are not required to perform safety functions are described in Section 7.7. In general, normal automatic operation of these control systems is assumed unless lack of operation would make the consequences of the incidents more adverse. In this case, the particular control system is assumed to be inoperative. No credit is taken in the analyses for any operator action prior to initiation of the event which would mitigate the consequences of the transient; however, the analyses are performed on the basis that the plant is being operated within all limiting conditions for operation (LCO) at the initiation of all events.

As stated, the automatic operation of the pressurizer level control system is not credited to perform any safety related function. The transient and accident analyses do assume the most adverse control of the pressurizer level control system to maximize the event consequences.

The charging system is safety related and is required to perform a post-accident function. UFSAR Section 9.3.4.3 (Safety Evaluation) describes that the capability of the charging system to borate is not compromised by stopping letdown flow. Because safe shutdown can be achieved without letdown flow, letdown flow does not have a post-accident function. For accidents which involve a safety injection actuation signal or containment isolation actuation signal, the letdown line is automatically isolated which is a safety function. This change assumes operation within the restrictions contained in the Technical Specifications, Core Operating Limits Report, and Technical Requirements Manual.

UFSAR Section 9.3.4.3.7 (Failure Mode and Effects Analysis) states at least one failure is postulated for each major component. This change does not change any of the physical structures, systems, or components. The pressurizer level control system was designed to allow manual control of the charging pumps and letdown control valves. UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) already described that automatic control is normally used during operation, but manual control may be used at any time. The proposed change also does not create any new physical system interactions that could cause a malfunction. Thus, the proposed change does not result in more than a minimal 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:

When determining which activities represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. Therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators. The UFSAR Chapter 15 accidents which could have their radiological consequences adversely impacted by charging/letdown flow were assessed to demonstrate the impact. The dose events that can potentially be impacted by increased charging/letdown flow are the UFSAR Section 15.1.2.4 (Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component), UFSAR Section 15.2.3.1.3 (Large Feedwater Line Break), UFSAR Section 15.6.3.1 (Primary Sample or Instrument Line Break), and UFSAR Section 15.6.3.2 (Steam Generator Tube Rupture).

These UFSAR events are potentially impacted because the steady state fuel activity release rate is a function of the amount of system cleanup that occurs due to letdown flow and activity decay. The larger charging/letdown flow and activity decay corresponds to a larger fuel activity release rate. For the accident induced iodine spiking doses, the fuel activity release rate is multiplied by a factor of 500 or 335. The maximum charging/letdown flow maximizes the fission products released to the reactor coolant system, which in turn maximizes the offsite doses [Reference 9 Attachment 1 Response 4]. UFSAR Section 15.1.2.4 (Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component), UFSAR Section 15.2.3.1.3 (Large Feedwater Line Break), UFSAR Section 15.6.3.1 (Primary Sample or Instrument Line Break), and UFSAR Section 15.6.3.2 (Steam Generator Tube Rupture) are all events which analyze an accident induced iodine spike.

UFSAR Section 15.6.3.1 (Primary Sample or Instrument Line Break) will be addressed first. Letter W3F1-2001-0088 [Reference 8] sent a revised letdown line break dose analysis to the NRC for approval which contained an iodine spiking curve based upon maximum charging and letdown flows. W3F1-2001-0088 Attachment 1 page 2 confirms this information as follows:

> Also, during the re-analysis it was determined that the pre-accident letdown flow assumed in the development of FSAR Figure 15.1-75 was non-conservative. The original curve assumed a letdown flow equal to approximately one charging pump. However, during periods of elevated RCS activity levels, letdown flow will be maximized for RCS cleanup in accordance with site off normal procedures. Therefore, two, or possibly, three, Charging Pumps may be in operation.

W3F1-2001-0088 Attachment 1 page 5 described to the NRC the impact of higher letdown flow on the accident induced iodine spike dose results as follows:

> The maximum letdown flow is 128 gpm. The new analysis used 144 gpm to conservatively bound the maximum letdown and identified/unidentified leakage. The maximum values are used because at elevated RCS activities site procedures direct letdown flow to be maximized to clean up the RCS. For the accident induced iodine spiking doses, the larger letdown flow produces more adverse consequences. The reason for the more adverse consequences is that the event is established from steady state conditions with the fuel activity release rate equal to the amount of cleanup that occurs due to letdown flow and activity decay. Thus, the larger letdown flow equates to a larger fuel activity release rate.

W3F1-2001-0088 Attachment 1 page 7 explained that the revised dose results remained within the 10CFR100 limits but exceeded the Standard Review Plan (SRP) limit for the accident induced iodine spiking results. The information was presented as follows:

> The new analysis results meet the SRP acceptance criteria with the exception of the EAB accident induced iodine spiking thyroid dose. The SRP acceptance criteria is a small fraction of the 10CFR100 limits (30 rem). The new analysis results are 70 rem which fall well within the 10CFR100 limits of 300 rem.

W3F1-2001-0088 Attachment 1 Section 4 described to the NRC the conservative assumptions used in the analysis as follows:

> A letdown line break starting from the most limiting parameters allowed by the TS LCO (1 pCi/gm) on RCS activity, pressure, temperature, primary to secondary leakage, and proceeding unmitigated for 30 minutes is highly unlikely. The additional use of conservative assumptions such as an iodine spiking factor of 500, maximum bounding letdown flow, worst case 95 percentile atmospheric dispersion factors, flashing fraction based on 560°F even though the break flow would travel through the regenerative heat exchanger and cool down, no activity plate out, no ground deposition, and no activity decay in the transit to the exclusion area boundary significantly increases the overall conservative nature of the calculation.

W3F1-2001-0088 Attachment 1 Section 4 further quantified to the NRC the conservative factors in the analysis and how the dose results could be a small fraction of 10CFR100 limits if any of the conservatisms were removed. This information is listed below:

> The new analysis accident induced iodine spiking results would remain below the SRP acceptance criteria if any one of the following normal plant operating parameters were used: RCS steady state activity, iodine spiking factor, letdown flow, or atmospheric dispersion factors. From the table above, the normal plant operating parameters demonstrate the values used in the analysis are conservative by a factor of 10 for RCS steady state activity, a factor of 2.5 for iodine spiking factor, about a factor of 3 for letdown flow, and about a factor of 11 for the atmospheric dispersion factors.
>
> Therefore, Waterford 3 proposes that due to the low occurrence probability of this event at the bounding conditions plus the conservative nature of the calculation that it is reasonable and appropriate to allow the accident induced iodine spiking consequences to fall well within the 10CFR100 limits (25% of Part 100 or 75 rem).

Letter W3F1-2002-0072 [Reference 9] describes communication between the NRC and Waterford 3 associated with the W3F1-2001-0088 submittal and responds with additional information to the NRC. W3F1-2002-0072 describes that the NRC explicitly understood that a

single charging pump would be used in the analysis (as the licensing basis) and the analysis contained sufficient conservatism to accommodate two or three charging pump operation during an operating cycle. This information is confirmed from W3F1-2002-0072 page 1 as follows:

> On May 10, 2002 Entergy and members of your staff participated in a conference call to discuss the number of charging pumps assumed to be in operation for the analysis. During this call, the NRC staff stated that an initial assumption of a single operating pump provides a suitable licensing basis analysis and has sufficient conservatism to accommodate two and three pump operating scenarios that may exist during the operating cycle.

W3F1-2002-0072 Attachment 1 Response 7 reiterated the discussion held with the NRC on the letdown line break analysis assumption for letdown/charging flow rates.

> During a May 10, 2002 conference call, the NRC staff stated it was acceptable to assume only one charging pump is in operation for this analysis therefore, 44 gpm is used to represent this plant configuration. Based on the May 10, 2002 conference call with the staff, it is Entergy's understanding that a single pump analysis provides a suitable licensing basis analysis and has sufficient conservatism to accommodate two and three pump operating scenarios that may exist during the operating cycle.

For the letdown line break event (UFSAR 15.6.3.1), the NRC safety evaluation report (SER) specifically addressed the charging/letdown required in the analysis in NRC Operating License Amendment 184 [Reference 10] Section 1.0 (Introduction) as follows:

> In Reference 1, the licensee assumed an initial condition of three charging pumps in operation to assure that the most severe radioactive releases would be considered. Because of this assumption, the Standard Review Plan (SRP) acceptance criterion of a small fraction (10 percent) of the 10 CFR Part 100 limits was exceeded and Nuclear Regulatory Commission (NRC) approval was required in accordance with 10 CFR 50.59. This application was noticed in the Federal Register on November 28, 2001 (66 FR 59504). After discussions with NRC, the licensee, in Reference 2, assumed an initial condition of a single operating charging pump which provides a suitable licensing basis and has sufficient conservatism to accommodate two and three-pump operating scenarios that may exist during the operating cycle. The results of this analysis fall within the acceptance criteria contained in the SRP, but the increase in dose exceeds 10 percent of the difference between the currently approved dose and the regulatory limit. Reference 2 was noticed in the Federal Register on October 29, 2002 (67 FR 66009), and superceded the biweekly Federal Register notice dated November 28, 2001 (66 FR 59504).

In addition, the NRC safety evaluation report for Operating License Amendment 184 Section 3.1.2.2 (Accident Induced Iodine Spiking) described that normally one charging pump is in operation which remains consistent with the LBDCR 16-0006 UFSAR Section 7.7.1.2.2 and 9.3.4 update. The NRC stated that the 3 charging pump operation is overly conservative for determining the equilibrium iodine appearance rate and the spiking model is overly conservative. As stated in W3F1-2001-0088 Attachment 1 Section 4 (above), normal plant operating parameters are conservative with respect to analysis inputs by a factor of 10 for reactor coolant system steady state activity, a factor of 2.5 for iodine spiking factor, and about a factor of 3 for letdown flow (with 3 charging pumps operating). This information was used by the NRC as a basis for determining that the charging pumps and spiking model had sufficient conservatisms to account for two or three charging pump operation during the cycle. The NRC Operating License Amendment 184 Section 3.1.2.2 pertinent information is as follows:

> Because during normal operation, Waterford 3 has only one pump in operation for

> letdown and the equilibrium iodine appearance rate is determined for normal operation, the staff's position is that the assumption of three pumps in operation for letdown is overly conservative for the determination of the equilibrium iodine appearance rate at Waterford 3. The staff communicated this position to the licensee in a telephone conference call on May 10, 2002, and by letter dated June 20, 2002, which was a request for additional information on the subject amendment request. During this conference call, the staff also informed the licensee that the staff was not apt to accept the dose consequences of the accident induced spiking not meeting the SRP acceptance criteria, considering the overly conservative nature of the licensee's spiking model.

The charging/letdown flow impact was specifically described to the NRC [Reference 8 and 9] and the NRC accepted the use of the normal charging/letdown flow in the letdown dose calculation due to the conservative assumptions already utilized in the analysis. The maximum charging and letdown flow impact is accounted for by the other analysis conservatisms and the NRC specifically approved this methodology [Reference 10]. The NRC specifically stated [Reference 10] that the initial condition of a single operating charging pump provides a suitable licensing basis and has sufficient conservatism to accommodate two and three pump operating scenarios that may exist during the operating cycle. This means that no analysis limitation exists preventing maximum charging and letdown. Thus, this change does not adversely impact the letdown line break dose analysis.

The UFSAR Section 15.2.3.1.3 (Large Feedwater Line Break) also evaluates an accident induced iodine spike. The event was submitted to the NRC during the conversion to alternate source term. Letter W3F1-2004-0053 Attachment 1 Section 1.2 [Reference 11] stated the following:

> The OC-MSLB/FWLB analysis bases the GIS spike on an assumed purification flow of 44 gpm, which corresponds to the normal operating plant condition of one charging pump operating. The acceptability of assuming one charging pump operating is discussed in the NRC Safety Evaluation Report (SER) concerning the letdown line break analysis. In the SER, the NRC noted that the assumption of three charging pumps operating is overly conservative for determining the equilibrium iodine appearance rate at Waterford 3. Therefore, the analysis assumed only one charging pump operating for determining the iodine appearance rate, which corresponds to normal plant operation.

The charging/letdown flow impact was specifically described to the NRC [Reference 11] and NRC approved this methodology in NRC Operating License Amendment 198 [Reference 14] for alternate source term. The maximum charging and letdown flow impact is accounted for by the other analysis conservatisms and the NRC specifically approved this methodology. This means that no analysis limitation exists preventing maximum charging and letdown. Thus, this change does not adversely impact the feedwater line break dose analysis.

The UFSAR Section 15.1.2.4 (Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component) also evaluates an accident induced iodine spike. The event was submitted to the NRC during the conversion to alternate source term. Letter W3F1-2004-0071 Attachment 1 Section 1.2 [Reference 12] stated the following:

> The IADVO will be documented/quantified in terms of the dose assessment for the Feedwater Line Break (FWLB) event. That is, the releases are assumed to be bounded by the releases associated with the analysis presented in Section 9.0 of W3F1-2004-0053.

The feedwater line break specifically addressed charging/letdown flow in W3F1-2004-0053 [Reference 11] Attachment 1 Section 1.2 which covers the inadvertent opening of an atmospheric dump valve event and the NRC approved this methodology in NRC Operating License Amendment 198 [Reference 14] for alternate source term. The maximum charging and letdown flow impact is accounted for by the other analysis conservatisms and the NRC specifically approved this methodology. This means that no analysis limitation exists preventing maximum charging and letdown. Thus, this change does not adversely impact the inadvertent opening of a steam generator atmospheric dump valve dose analysis.

The UFSAR Section 15.6.3.2 (Steam Generator Tube Rupture) also evaluates an accident induced iodine spike. The event was submitted to the NRC during the conversion to alternate source term. Letter W3F1-2004-0101 Attachment 1 Section 5.1.1 [Reference 13] stated the following:

> An accident GIS case, where the primary system transient associated with the SGTR causes an iodine spike in the primary system. A spiking factor of 335 is specified per Appendix F to RG 1.183. Consistent with the Letdown Line Break analysis for Waterford 3, flow from one charging pump is assumed for steady-state operating conditions prior to the SGTR event for the purposes of determining the iodine appearance rate for the GIS case.

The charging/letdown flow impact was specifically described to the NRC [Reference 13] and NRC approved this methodology in NRC Operating License Amendment 198 [Reference 14] for alternate source term. The maximum charging and letdown flow impact is accounted for by the other analysis conservatisms and the NRC specifically approved this methodology. This means that no analysis limitation exists preventing maximum charging and letdown. Thus, this change does not adversely impact the steam generator tube rupture dose analysis.

The accidents previously evaluated in the UFSAR were assessed for potential impacts and no adverse consequences were identified. The proposed change does not adversely impact the capability of charging or letdown to perform their specified functions. Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

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

BASIS:

When determining which malfunctions represent "more than a minimal increase in consequences" pursuant to 10 CFR 50.59, it must be recognized that "consequences" means dose. This change addresses updating UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) and Section 9.3.4 (Chemical and Volume Control System) to resolve the CR-WF3-2016-0243 identified conflict and to make USFAR Section 9.3.4 consistent with UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) which states automatic control is normally used during operation, but manual control may be used at any time. This change will allow manual control of the pressurizer level control system for more than just increased purification flow. The normal method of controlling pressurizer level in automatic is still preferred and is not changed by this update.

In general, design basis accidents have an event initiator and may assume one active single failure. The design basis accidents have identified their limiting single failures with respect to a

specific acceptance criteria, the potential operator error would be no different than the active single failure of pressurizer level control instrument. UFSAR Table 9.3-15 (Chemical and Volume Control System Failure Mode and Effects Analysis) shows a failure mode and effects analysis for the CVCS. At least one failure is postulated for each major component.

This change does not change any of the physical structures, systems, or components. The pressurizer level control system was designed to allow manual control of the charging and letdown control valves. UFSAR Section 7.7.1.2.2 (Pressurizer Level Control System) already described that automatic control is normally used during operation, but manual control may be used at any time. The proposed change also does not create any new system interactions that could cause a malfunction.

Therefore, the pressurizer level control manual operation is no more adverse than that already analyzed in the UFSAR. The proposed change also does not create any new system interactions that could cause a malfunction. The proposed change does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

BASIS:

This change is not a new safety related operator action because the pressurizer level control system has no safety related credited function as specifically described in UFSAR Section 7.7 and NUREG-0787 Section 7.7. UFSAR Section 15.0.2 states the following:

> Systems which are not required to perform safety functions are described in Section 7.7. In general, normal automatic operation of these control systems is assumed unless lack of operation would make the consequences of the incidents more adverse. In this case, the particular control system is assumed to be inoperative. No credit is taken in the analyses for any operator action prior to initiation of the event which would mitigate the consequences of the transient; however, the analyses are performed on the basis that the plant is being operated within all limiting conditions for operation (LCO) at the initiation of all events.

As stated, the automatic operation of the pressurizer level control system is not credited to perform any safety related function. The transient and accident analyses do assume the most adverse control of the pressurizer level control system to maximize the event consequences. The pressurizer level control system are evaluated as accident initiators and single failures. The transient and accident analyses that are already included in the UFSAR are as follows:

- UFSAR Appendix 5.2B (Low Temperature Overpressure Protection During Heatup and Cold Shutdown) demonstrates that the shutdown cooling system overpressure protection meets the low temperature overpressure protection criteria by analyzing the limiting transients.
- UFSAR Section 15.2.1.3 (Loss of Condenser Vacuum) limiting single failure is associated with the pressurizer level control.
- UFSAR Section 15.2.2.5 (Loss of Normal Feedwater Flow) limiting single failure is associated with the pressurizer level control.
- UFSAR Section 15.4.1.5 (CVCS Malfunction - Inadvertent Boron Dilution) is a malfunction which results in unborated water being pumped at the maximum possible rate into the RCS by the demineralized water supply system is assumed

to occur. For this to occur, one or more charging pumps must be on, the primary water makeup water pumps must be on, and the demineralized water supply system must be aligned to supply water to the charging pump suction via the volume control tank. Since at least three simultaneous equipment malfunctions would be required to produce the above conditions, the incident could only be the result of improper operator action accompanied by a single equipment malfunction.

UFSAR Section 15.5.1.1 (Chemical and Volume Control System Malfunction) addresses an accident caused by the startup of all backup charging pumps with letdown flow at minimum.

UFSAR Section 15.5.2.1 (Chemical and Volume Control System Malfunction with a Concurrent Single Failure of an Active Component) addresses an accident caused by the startup of all backup charging pumps with letdown flow isolated.

These events all involve increased reactor coolant system inventory and loss of shutdown margin if the water is unborated. For any event which causes a loss of reactor coolant inventory (such as charging isolated with full letdown flow) would be bounded by the loss of coolant accidents (UFSAR Section 15.6.3.3).

Based upon a review of the UFSAR events, the initiators and single failures bound the manual operation of the pressurizer level control system failure modes. Thus, an accident of a different type than previously evaluated 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:
This change impacts the automatic operation of the pressurizer level control system which has no safety related credited function as specifically described in UFSAR Section 7.7 and NUREG-0787 Section 7.7.

NUREG-0787 Supplement 1 [Reference 6] Section 7.7 (Control Systems Not Required for Safety) states the following:

> With regard to the effects of control system failures or malfunctions, the analyses reported in Chapter 15 of the Final Safety Analysis Report are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents, including those related to control systems. Based on the conservative assumptions' made in defining these "design bases" events and on our detailed review of the analyses, it is likely that these analyses adequately bound the consequences of single control system failures. To provide assurance that the Chapter 15 analyses adequately bound events initiated by a single credible failure or malfunction, the staff requires that a review be conducted to identify any power sources or sensors which provide power or signals to two or more control systems, and to demonstrate that failures or malfunctions of these power sources or sensors will not result in consequences outside the bounds of the Chapter 15 analyses or beyond the capability of operations of safety systems.

The NUREG-0787 Supplement 1 validation that UFSAR Chapter 15 events bounded non-safety control system failures was addressed in NUREG-0787 Supplement 8 [Reference 7]. NUREG-0787 Supplement 8 Section 7.7.6 (Single Failure of Control System Study) states the following:

> In response to IE Bulletin 79-22, the applicant addressed consequential control system failures following a high energy line break (HELB) and the staff found the applicant's evaluation acceptable in the SER. Regarding IE Bulletin 79-27, which addresses loss of power to control systems, the applicant, in letters dated February 27 and May 7, 1984, provided information summarizing failure modes and effects analysis. The analysis indicated that no case was found where design modifications were required to ensure the ability to achieve a cold shutdown condition. Subsequently, in Supplement 6, the staff concluded that the issue was considered closed.

UFSAR Table 9.3-15 (Chemical and Volume Control System Failure Mode and Effects Analysis) shows a failure mode and effects analysis for the CVCS. This change does not create a failure not already bounded by the information in the table.

UFSAR Section 15.2.1.3 (Loss of Condenser Vacuum) limiting single failure is associated with the pressurizer level control. The UFSAR specifically details the failure as follows:

> The single failure considered for the peak primary pressure cases was the failure of the pressurizer level measurement channel. The effect of that failure is a false low pressurizer level signal, which would result in the activation of the charging pumps and the closing of the letdown valves.
>
> Analyses were also conducted to demonstrate that adequate operator action times exist to prevent the pressurizer from becoming liquid filled for this event. A failure in the Pressurizer Level Control System (PLCS) was assumed that resulted in terminating letdown flow and starting all three charging pumps. Operator action at 15 minutes into the event to trip charging pumps and, if applicable, restore letdown demonstrated that margin exists to a solid pressurizer condition. This time for operator action is greater than the 10 minutes allowed per NUREG-0800.
> (This means the operator has more time available than the minimum requirement per NUREG-0800.)

UFSAR 15.2.2.5 (Loss of Normal Feedwater Flow) limiting single failure is associated with the pressurizer level control. The UFSAR specifically details the failure as follows:

> Analyses were also conducted to demonstrate that adequate operator action times exist to prevent the pressurizer from becoming liquid filled for this event. A failure in the Pressurizer Level Control System (PLCS) was assumed that resulted in terminating letdown flow and starting all three charging pumps. Operation action at 15 minutes into the event to trip charging pumps and, if applicable, restored letdown demonstrated margin exists to a solid pressurizer condition. This time for operator action is greater than the 10 minutes allowed per NUREG-0800.
> (This means the operator has more time available than the minimum requirement per NUREG-0800.)

As stated in NUREG-0787, the UFSAR Chapter 15 analyses were used to demonstrate no adverse impact associated with control system not required with safety. The analysis assumed failure of the pressure level control system and limiting single failures and demonstrate acceptable transient results. This change remains bounded by the UFSAR analyses.

In addition, the pressurizer level control system is not required for safety, 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 does not exist.

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 potentially impacted fission product barriers are associated with the reactor coolant system pressure boundary and associated interfacing pressure boundary (shutdown cooling). These potential malfunctions could produce an unplanned increase in reactor coolant inventory caused by an equipment or electrical malfunction or by an operator error which starts one or more charging pumps and interrupts letdown flow [UFSAR Section 15.5.1.1]. UFSAR Appendix 5.2B (Low Temperature Overpressure Protection During Heatup and Cold Shutdown), UFSAR Section 15.5.1.1 (Chemical and Volume Control System Malfunction), and UFSAR Section 15.5.2.1 (Chemical and Volume Control System Malfunction with a Concurrent Single Failure of an Active Component) provide the analyses of these events.

UFSAR Appendix 5.2B (Low Temperature Overpressure Protection During Heatup and Cold Shutdown) demonstrates that the shutdown cooling system overpressure protection meets the low temperature overpressure protection criteria by analyzing the limiting transients. The most limiting transients initiated by a single operator error or equipment failure are (1) an inadvertent safety injection actuation (mass input) and (2) a reactor coolant pump start when a positive steam generator to reactor vessel ΔT exists (energy input). The inadvertent safety injection is applicable to this change. The inadvertent safety injection could cause both charging pumps to start and isolate letdown flow. The analysis demonstrated that the shutdown cooling relief valves prevent overpressurization. This change would remain bounded by this analysis and would not adverse impact its conclusions.

UFSAR Section 15.5.1.1 (Chemical and Volume Control System Malfunction) addresses an accident caused by the startup of all backup charging pumps with letdown flow at minimum. The maximum primary and secondary pressures are less than 110 percent of the design pressure following a CVCS malfunction, thus assuring that the integrity of the reactor coolant system and main steam System is maintained. This change would remain bounded by this analysis and would not adverse impact its conclusions.

UFSAR Section 15.5.2.1 (Chemical and Volume Control System Malfunction with a Concurrent Single Failure of an Active Component) addresses an accident caused by the startup of all backup charging pumps with letdown flow isolated. The maximum primary and secondary pressures are less than 110 percent of the design pressure following a CVCS malfunction, thus assuring that the integrity of the reactor coolant system and main steam System is maintained. This change would remain bounded by this analysis and would not adverse impact its conclusions.

As stated in NUREG-0787 Supplement 1 [Reference 6] Section 7.7 (Control Systems Not Required for Safety), the transient and accident analysis bounds the potential failures associated with the pressurizer level control system. 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:

Methods of evaluation means the calculational framework used for evaluating behavior or response of the facility or an SSC [Reference 4 Section 3.10]. This change is not impacting the methodology or topical reports used in the UFSAR analyses. UFSAR Section 15.0.2 states the following:

> Systems which are not required to perform safety functions are described in Section 7.7. In general, normal automatic operation of these control systems is assumed unless lack of operation would make the consequences of the incidents more adverse. In this case, the particular control system is assumed to be inoperative. No credit is taken in the analyses for any operator action prior to initiation of the event which would mitigate the consequences of the transient; however, the analyses are performed on the basis that the plant is being operated within all limiting conditions for operation (LCO) at the initiation of all events.

As stated, the automatic operation of the pressurizer level control system is not credited to perform any safety related function. The transient and accident analyses do assume the most adverse control of the pressurizer level control system to maximize the event consequences. The existing UFSAR evaluations continue to follow this methodology. 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.