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W3F1-2004-0110

November 29, 2004

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

Subject: Report of Facility Changes, Tests and Experiments and Commitment Changes
Waterford Steam Electric Station, Unit 3 (Waterford 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). This report covers the period from June 1, 2003 through May 31, 2004. This submittal also includes a CD-ROM of the 10CFR50.59 Evaluation for each change and a summary report of Commitment Changes for the same time period.

If you have any questions regarding this report, please contact Michael E. Mason at (504) 739-6673.

There are no new commitments contained in this submittal.

Sincerely,

R.J. Murillo
Acting Licensing Manager

RJM/MEM/cbh
Attachment(s)

IE47

cc: Dr. Bruce S. Mallett
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Attachment 1

W3F1-2004-0110

Waterford 3 10 CFR 50.59 Summary Report

10CFR50.59 Evaluation Number	Initiating Document	Summary
02-001	ER-W3-2001-1211-000	Pressurizer Heater Leak Rework
03-001	ER-W3-2002-0335-000	Low Pressure Safety Injection Containment Isolation Valve Bypass Line Removal
03-006	ER-W3-2002-0585-000	Component Cooling Water Containment Isolation Valves Closing Time Adjustment
03-007	ER-W3-2003-0257-000	Technical Specification Bases 3.6.3, Technical Requirements Manual 3.6.3 and Technical Requirements Manual Bases 3.6.3
03-008	ER-W3-2003-0112-000	Technical Specification Bases 3/4.5.2
03-010	ER-W3-2002-0675-000	Boric Acid Makeup Tank Shutdown Boric Acid Concentration
03-011	ER-W3-2002-0283-002	Shutdown Cooling Containment Isolation Valves Leakage Testing Category
03-012	STI-W3-2003-005	Steam Generator Chemical Cleaning
03-013-1	ER-W3-2003-0585-000	In Core Instrumentation and Fuel Assembly Misloading Analysis and TRM 3.3.3.2
03-014	NF-03-W-WTFD-42 and NF-03-W-WTFD-40	CYCLE 13 Core Operating Limits Report
03-015	ER-W3-2003-0681-000	Repair of Broken In Core Instrumentation Thimble T04
03-017	ER-W3-2003-0117-000	Steam Generator Tube Plugging Margin Limit
03-035	STI-W3-2003-005	Steam Generator Chemical Cleaning Fuel Handing Building Heating Ventilation and Air Conditioning Impact
03-043	ER-W3-2003-0135	Irradiation-Induced Growth Of In Core Instrumentation Thimbles
04-001	ER-W3-2000-0106-002	Reactor Coolant Pumps Controlled Bleed Off Design Pressure Revision
04-002	ER-W3-2003-0755	Steam Bypass Control System Response Impacting Reactor Trip Between 50% And 70% Power
04-003	STI 04-WF3-0002-00	Control Room Tracer Gas Test

Facility: Waterford-3

Document Reviewed: ER-W3-2001-1211-000

System Designator(s): RCS

Check the applicable review(s):

☐	SCREENING	Sections I, II, and III required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, and IV required
☒	50.59 EVALUATION Evaluation #: 02-001	Sections I, II, III, and V required

NOTE: Only the sections required as indicated above must be included in the Review.

I. SIGNATURES / OVERVIEW

Preparer: JB Perez/EOI/Design Engineering/01-21-02
Signature / Name (print) / Company / Department / Date

Reviewer: Richard T. Finch/EOI/Design Engineering/01-21-02
Signature / Name (print) / Company / Department / Date

(PSRC): 1-22-01
Chairman's Signature / Date (N/A for Screenings and 50.59 Evaluation Exemptions)

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:

Description of Proposed Change

This ER will authorize installation of mechanical nozzle seal assemblies (MNSA-2) on the side shell and bottom head of the pressurizer as an alternate to weld repair to restore the Reactor Coolant System (RCS) pressure boundary as a result of instrument nozzle or heater sleeve leaks. The MNSA-2 clamps may also be installed proactively before leaks occur. The Safety Evaluation addresses 1) the installation of the clamps and operation with the MNSA clamps installed to the limits of Technical Specification 3.4.9, and 2) the impact on instrumentation and system interface piping for all modes of operation with the MNSA-2 clamps installed. The scope of the change includes two pressurizer instrument nozzles on the bottom head, one temperature nozzle on the side shell, and 29 pressurizer heater sleeves (one of the 30 heater sleeves was plugged in Refuel 10). This Safety Evaluation does not authorize final approval of the MNSA-2 clamps to restore RCS integrity. Final approval of the alternate repair to restore the RCS integrity is contingent upon NRC approval.

The repair will consist of counter boring a hole in the pressurizer base material to provide a flat, perpendicular surface and attaching a stainless steel mechanical restraining device over the leaking instrument nozzle or heater sleeve. Installation includes a grafoil seal arrangement for the gap between the nozzle and vessel penetration, similar to the concept of a valve stem packing arrangement. This restraining device is called the RCS mechanical nozzle seal assembly (MNSA-2) which replaces the function of the "J" weld between an Inconel 600 instrument nozzle and the pressurizer, to prevent leakage from cracks caused by Primary Water Stress Corrosion Cracking. The design of the MNSA-2 relocates the pressure boundary from the nozzle or heater sleeve attachment weld on the inside surface of the vessel to a Grafoil seal located in a new counterbore machined in the external surface of the pressurizer. It also acts to restrain the instrument nozzle or heater sleeve from ejecting if the "J" weld completely fails (360 degree circumferential crack). The MNSA-2 meets all applicable design requirements for the RCS and can be installed without offloading the core and with little or no outage critical path impact. The MNSA-2 is a second generation design which improves on the original MNSA design that was installed on the hot legs during RF-9 and removed RF-10.

II. SCREENING

A. Licensing Basis Document Review

Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒		
TS	☐	☒		
NRC Orders	☐	☒	☐	
If "YES", obtain NRC approval prior to implementing the change. (See Section 5.1.13 for exceptions.)				

LBDs controlled under 50.59	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
UFSAR	☒	☐		See "BASIS" in Section C, below
TS Bases	☐	☒		
Technical Requirements Manual	☐	☒		
Core Operating Limits Report	☐	☒		
Fire Hazard Analysis	☐	☒	☐	
Fire Protection Program	☐	☒	☐	
Offsite Dose Calculations Manual	☐	☒	☐	
Process Control Program	☐	☒	☐	
NRC Safety Evaluation Reports[1]	☐	☒		
If "YES", perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.				

LBDs controlled under 72.48	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Cask UFSAR	☐	☐	☒	
Certificate of Compliance	☐	☐	☒	
If "YES", evaluate/process any changes in accordance with 72.48				

LBDs controlled under other regulations	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒		
Emergency Plan[2]	☐	☒		
Security Plan[2, 3]	☐	☒		
Inservice Inspection Program[4]	☐	☒	☐	
Inservice Testing Program[4]	☐	☒	☐	
If "YES", evaluate/process any changes in accordance with the appropriate regulation.				

[1] If "YES," see Section 5.1.5.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed.
[3] The Security Plan is classified as safeguards and can only be reviewed by personnel with the appropriate security clearance. The Preparer should notify the security department of potential changes to the Security Plan.
[4] If "YES", process the change in accordance with the 10CFR50.55a control program.

B. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ Yes ☒ No — If "yes," perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.

C. Basis

(Provide a basis for the "no" items checked in Sections II.A and II.B, above. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. If a 50.59 Evaluation is required, this section may be N/A'd.)

The use of MNSA-2 clamps is not described in the FSAR, however a Safety Evaluation is being performed because the proposed change provides an alternate repair method which may not be completely addressed by the ASME Boiler and Pressure Vessel Code. The change is being submitted to the NRC for approval. In the past, Waterford-3 and other utilities have requested NRC approval for the use of the original MNSA to repair leaking RCS nozzles. The use of the original MNSA is described in the FSAR (Table 5.2.3).

D. Is the validity of this Review dependent on any other change? *(See Section 5.2.2.4 of the EOI 10CFR50.59 Program Review Guidelines)* ■ Yes ☐ No

If "Yes," list the required changes.

_EC-M94-011 Stress Analysis for Pressurizer

_SQ-MN-347 Seismic Qualification for MNSA

_PWR-R&R-001 Relief Request for MNSA-2

E. References

[Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.2.2.4 of LI-101.]

Documents:
ER-W3-99-0198-00
Docket # 50-382 SER for MNSA
Ltr. W3F1-99-0043, (MNSA for hot leg)
DS-ME-01-1, R0 "Westinghouse Design Specification for MNSA-2"
DAR-ME-01-2, R0 "Comparision of MNSA and MNSA-2"

Keywords:

MNSA, mechanical nozzle, seal assembly

FSAR Sections Reviewed:

FSAR Sections 5.2.1, 5.4.3, 5.4.10

Table 5.2-3, 5.4-6

FSAR Figures Reviewed:

Fig. 5.1-3

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations."

Will the proposed Change being evaluated:

Yes	No	
☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
☐	☒	Involve dredging activities in a lake, river, pond, or stream?
☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
☐	☒	Discharge any chemicals new or different from that previously discharged?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?
☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?
☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
☐	☒	Involve the installation or modification of a stationary or mobile tank?
☐	☒	Involve the use or storage of oils or chemicals?
☐	☒	Involve burial or placement of any solid wastes in the site area that may effect runoff, surface water, or groundwater?

V. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after PSRC approval, if available.)

Brief description of change, test, or experiment:

This ER authorizes installation of mechanical nozzle seal assembly (MNSA-2) clamps to restore the pressure boundary of the pressurizer in the event of instrument nozzle or heater sleeve leaks. The MNSA-2 clamps may also be installed proactively before leaks occur. The change may be applied to either or both of the two instrument nozzles on the bottom head, the one Resistance Temperature Detector (RTD) nozzle on the side shell or the twenty nine heater sleeves. This Safety Evaluation does not authorize final approval of the MNSA-2 clamps to restore RCS integrity. Final approval of the alternate repair to restore the RCS pressure boundary is contingent upon NRC approval.

The repair will consist of machining a counterbore around the nozzle or heater sleeve and attaching a stainless steel mechanical restraining device over the leaking instrument nozzle or heater sleeve with bolts threaded into 4 holes drilled and tapped in the pressurizer base material. Installation includes a grafoil seal for the gap between the nozzle and vessel penetration, similar to the concept of a valve stem packing arrangement. The MNSA-2 will replace the function of the "J" weld between the nozzle and the pressurizer to prevent leakage and to restrain the instrument nozzle or heater sleeve from ejecting if the "J" weld completely fails.

Reason for proposed Change:

Primary water stress corrosion cracks (PWSCC) of Inconel (Alloy 600) penetrations in the RCS has become a significant problem in PWRs over the last ten years. These penetrations involve nozzles which are inserted through an opening in the vessel wall and are welded to the inside of the vessel wall by a J-groove weld. PWSCC has been found in pressurizer heater sleeves, pressurizer instrument nozzles, hot leg nozzles and Control Element Drive Mechanism (CEDM) nozzles. Recently (2001), PWSCC has been found on the CEDM nozzles and attachment "J" welds at Oconee 1, 2 and 3 and at ANO-1.

During Refuel 9, RCS leakage was found on two instrument nozzles on the pressurizer top head. Three hot leg nozzles were also leaking and were repaired by using MNSA clamps. Leakage was also found on a pressurizer heater sleeve during the last outage (RF-10) and the sleeve was repaired by plugging.

Leaking instrument nozzles or heater sleeves on the bottom of the pressurizer are difficult to repair because of their location. Weld repair of these nozzles would require a long drain down window or core off load with an outage critical path impact of at least 6 days. Since they can be installed without drain down of the pressurizer, with significantly less dose than a welded repair, MNSA-2 clamps are more desirable for these repairs. The MNSA-2 is an improved design over the original MNSA clamps. Use of the MNSA-2 will be approved by the NRC prior to operation above Technical Specification 3/4.4.9 limits (RCS Temperature of 286 Deg F per ER-W3-99-0198-001).

50.59 Evaluation summary and conclusions

This evaluation concludes that the proposed installation of the MNSA-2 will not degrade the integrity of the pressurizer or any other RCS pressure boundary. There will be no impact on

instrumentation and system interface piping for all modes of operation with the MNSA-2 clamps installed. All changes are within the Reactor Containment Building, and there are no new system interactions created. There is no effect on nuclear safety and this change does not require any Technical Specification change.

B. License Amendment Determination

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 FSAR?

BASIS:

The only accidents that are potentially affected by this modification is the small break LOCA and loss of shutdown cooling.

Installation

The MNSA-2 will be installed in mode 5 or 6 while on shutdown cooling. The MNSA-2 installation process is non-intrusive on the existing nozzle or heater sleeve pressure boundary integrity. Nozzle ejection during installation (mode 5 or 6) is not a concern recognizing that the only stresses that exist during this evolution are those created by pressurizer head pressure, insignificant loads associated with field machining, the dead weight of the nozzle and attached pipe, thermal loads, and potential seismic loads.

The MNSA-2s are attached to the pressurizer with the SA-453 Grade 660 bolts threaded into four tapped holes, arranged in a circular pattern around the nozzle. The addition of the holes in the pressurizer is being analyzed and will be documented in the Stress Report. The analysis is being performed to the requirements of ASME Section III, 1971 Edition through and including the Summer 1971 Addenda. The analysis will address fatigue to demonstrate that the Code prescribed cumulative usage factor of 1.0 is not exceeded, that there is adequate reinforcement in the wall of the pressurizer for the bolt holes, and that the stresses do not exceed the allowables as stated in the Code. The Stress Report changes will be approved prior to returning the system to Operations (Controlled by RTS Action established in the ER Database).

The design, materials, fabrication, examination and testing of the mechanical nozzle seal assembly meet Class 1 requirements of ASME III, 1989 Edition, no addenda, in accordance with Design Specification DS-ME-01-1. Installation of the MNSA-2 will meet ASME XI requirements, consistent with Waterford 3 ASME XI program per 10CFR50.55a. The design of the MNSA-2 is in accordance with NB-3671.7 and the rules of NB-3200. Prototype testing is being performed to seismically qualify the MNSA-2 design. Additional testing will be performed to validate the design. Installation dimensional clearances between the pressurizer and mechanical nozzle seal assembly top plate will be provided to allow for differential thermal expansion, which will not create significant thermal stresses on the existing nozzle, thermowell or associated system piping

The MNSA-2 has advantages over the original MNSA that was approved by the NRC for one operating cycle on the Hot Leg piping. The MNSA-2 utilizes a counterbore on the pressurizer material to provide a machined surface perpendicular to the axis of the nozzle. The previous MNSA clamp was installed on the rough surface of the piping on a curved surface. The flat, smooth surface on the MNSA-2 design ensures a more effective seal. The MNSA-2 also includes live-loaded bolting to prevent loosening of the joint and a leak-off line to channel any leakage away from the attachment bolts.

Final Design

The MNSA-2 clamps fit around the outside of the nozzles or sleeves and the configuration, instrumentation and circuits (side shell temperature nozzle) are not affected by this repair. As such, this repair will not impact obtaining of pressurizer temperatures and level readings or RCS performance. The change in insulation around the MNSA-2 and side shell RTD, if required, will not affect the accuracy or qualification of the RTD.

Installation of the MNSA-2s will place minimal additional loads on piping attached to the pressurizer that will be within the structural capability of the piping.

Therefore, the proposed activity will not increase the frequency of occurrence of an accident evaluated previously in the safety analysis report.

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 FSAR?** ☐ Yes ☒ No

BASIS:

Installation

Operation of the pressurizer heaters, the side shell RTD or the bottom head level instrumentation is not required during modes 5 and 6. The temperature indication from the RTD is not safety-related.

As discussed in question 1, analysis shows that there is adequate reinforcement in the wall of the pressurizer for the bolt holes. Therefore, inventory will not be lost which could result in loss of shutdown cooling as a result of this modification.

Final Design

The insulation configuration to maintain environmental qualification of the side shell RTD and heater sleeves will be modified for MNSA-2 installation. The insulation will be the same type as the current insulation (NUKON) but the configuration will allow installation of MNSA-2 clamps and facilitate future visual inspections of the nozzles.

This repair meets seismic category 1 requirements and will not impose unacceptable loads on the RCS. Installation of the MNSA-2s will place minimal additional loads on the piping attached to the pressurizer instrument nozzles.

This repair has no affect on system protection features, or the support systems for its equipment. Also, this repair will not increase the frequency of operation of system equipment or impose more severe testing requirements on systems or equipment.

Therefore, the proposed activity will not increase the likelihood of occurrence of a malfunction of structures, systems or components important to safety evaluated previously in the safety analysis report.

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

BASIS:

Installation

The bottom head level instruments, RC ILT0103 and RC ILT1103A provide a means for monitoring and controlling pressurizer level during plant operation. The side shell RTD, RC ITE-0101, provides a means for measuring temperature and the heaters are used for control of RCS pressure during plant operations. These functions are not required during Modes 5 and 6.

Final Design

Installation of the mechanical nozzle seal assemblies will restrain the nozzles in place. As such, instrumentation and circuits would not be affected by this repair method. In addition, in the unlikely event that the seal (packing material) fails and causes RCS pressure boundary leakage, UFSAR Chapter 15, Small Break Loss of Coolant Accident (SBLOCA) Accident Analysis, would bound the event. As discussed in question 1, this repair meets seismic category 1 requirements and the installation of the MNSA-2s will have minimal impact on the integrity of the pressurizer.

This repair has no affect on system protection features, or the support systems for its equipment. Redundancy will ensure an operational protection system should an instrument nozzle fail rendering an instrument inoperable. Also, this repair will not increase the frequency of operation of system equipment or impose more severe testing requirements on systems or equipment.

The MNSA-2 is installed around an instrument nozzle or heater sleeve. A seal is created by compressing the grafoil split packing at the nozzle and the pressurizer OD interface. There is no affect on the heaters, the level control nozzles or the side shell RTD. The change in the insulation configuration will have no affect on the RTD terminal block temperature and will not have any affect on the instrument accuracy or qualification. The temperature reading from the side shell RTD is not safety related.

Therefore, the proposed change will not adversely impact any of 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 FSAR?**

BASIS:

☐ Yes
☒ No

Systems and components important to safety that could be affected by this modification are the reactor coolant pressure boundary, pressurizer level and temperature instrumentation, and the heater sleeves.

Installation

Pressurizer heaters, the side shell RTD and the pressurizer level instruments are not required in the shutdown modes.

As discussed in question 1, analysis will show that there is adequate reinforcement in the wall of the pressurizer for the bolt holes and the counterbore Therefore, inventory will not be lost which could result in loss of shutdown cooling as a result of this modification.

Final Design

As discussed in questions 1 and 3, there will be minimal additional loads imposed on the nozzles and attached system piping as a result of the installation of the MNSA-2s. The additional loads will be within the structural capability of the pressurizer. The consequences of failure of a nozzle with the MNSA-2 installed are the same as the failure of a nozzle without the MNSA-2, i.e., a small break LOCA. Evidence shows that a catastrophic failure of an Inconel nozzle is not expected to occur. However, in the event that this type of failure does occur, the consequences will not impact the safety of the plant. Specifically, this type of accident has been previously evaluated in the UFSAR and a postulated failure of one RCS instrument nozzle is bounded by this analysis.

Therefore, the proposed change does not increase the consequences of a malfunction of a system, structure, or component important to safety previously evaluated in the SAR.

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

BASIS:

☐ Yes
☒ No

Accidents considered for this repair method are a loss of coolant accident caused by failure of the instrument nozzle, heater sleeves, and failure of the mechanical nozzle seal assembly. These types of failures (i.e. instrument nozzle failures) have been previously considered and analyzed in Chapter 15. There are no other known failures that could occur.

Therefore, the proposed activity will not create the possibility of an accident of a different type than any evaluated previously in the safety analysis report.

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 FSAR?**

BASIS

☐ Yes
☒ No

Installation

The drilled holes in the pressurizer have the potential for reducing the structural integrity of the RCS pressure boundary. As stated previously, a calculation is being prepared that analyzes the bolt holes in the pressurizer

Final Design

The pressurizer configuration is not changed as a result of the addition of the MNSA-2 clamp, therefore, there will not be any impact due to flood, missile and wind (reference Design Basis Document DBD-009 section 3.1.1.3.3 through 3.1.1.3.5). Since the MNSA-2s are designed and installed as a threaded/bolted fixture; it is considered a multiple bolt attachment similar to a valve bolted bonnet. Calculated stresses have been reviewed and compared to the ultimate strength of the material. The stresses resulting from failure of a single bolt or tie rod will not be greater than the ultimate strength of the remaining bolts/tie rods. Therefore, a single failure of a bolt or tie rod will not create a new missile hazard or any other hazard. The results of failure of a MNSA-2 will be the same as the failure results of an RCS nozzle.

Therefore, the proposed activity will not create the possibility of a malfunction of systems or components important to safety with a different result than any evaluated previously in the safety analysis report

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

BASIS:

☐ Yes
☒ No

Installation

Drilling into the pressurizer base material is controlled by installation procedures and a stop is provided on the drilling apparatus to assure the bolt hole depths are within allowables. Torquing of the MNSA-2 bolts into the pressurizer will be performed at temperatures above the Reference Nil Ductility Temperature (RT_{NDT}) of 30^0 F to ensure that the bolting stress does not create a potential for brittle failure.

Final Design

The limit for a fission product barrier, the reactor coolant pressure boundary, is not reduced because 1) the analysis will show that there is adequate reinforcement in the pressurizer wall for the bolt holes and that the stresses do not exceed the allowables as stated in the Code, 2) there is no impact to the pressurizer heaters or instrumentation , and 3) the loads on the attached instrument piping will change negligibly and will be within the structural capability of the piping.

Therefore, the proposed activity does not reduce or alter any fission product barrier as described in the FSAR.

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

BASIS:

The ER does not involve a change in any methods of evaluation. The change does not affect any calculations, computer codes, procedures, or any other methods of evaluating design bases information.

☐ Yes
☒ No

I. OVERVIEW / SIGNATURES

Facility: Waterford 3

Document Reviewed: ER-WF3-2002-0335-000 **Change/Rev.** 0

System Designator(s)/Description: SI / Safety Injection

Description of Proposed Change

CR-WF3-2002-0818 identified that voids are repeatedly forming in the Low Pressure Safety Injection piping at the containment building penetrations. The cause of the voiding at Reactor Containment Building Penetrations 37 and 39 has been potentially attributed to leakage from bypass valves SI-1421A and B, which allow nitrogen-saturated water from the Safety Injection Tank to enter the lower pressure piping upstream of valve SI-142A or B, where the nitrogen comes out of solution and results in system voiding. Removal of the bypass line and replacement with an independent drain and test connection will eliminate the leak path and not impact any design function of the Safety Injection system.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).
- ☐ The proposed activity is controlled by another regulation per Section 5.2.2.3.

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, and III required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, and IV required
☒	50.59 EVALUATION (#: 03-001)	Sections I, II, III, and V required

Preparer: Thomas R. Hempel Thomas R. Hempel / EOI / DE / 1-23-03
Name (print) / Signature / Company / Department / Date

Reviewer: Gary E. Payne Gary E. Payne / EOI / DE / 1·23·03
Name (print) / Signature / Company / Department / Date

OSRC: K. Peters K. Peters 1/29/03
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents? (Check "N/A" for those documents that are not applicable to the facility.)**

Operating License	YES	NO	N/A	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒		
TS	☐	☒		
NRC Orders	☐	☒	☐	
If "YES", obtain NRC approval prior to implementing the change. (See Section 5.1.13 for exceptions.)				

LBDs controlled under 50.59	YES	NO	N/A	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐		FSAR Figure 6.3-1 Sheet 2 (by reference) DRN 02-1868 FSAR Figure 6.3-1 Sheet 4 DRN 02-1896
TS Bases	☐	☒		
Technical Requirements Manual	☐	☒		
Core Operating Limits Report	☐	☒		
Offsite Dose Calculations Manual	☐	☒		
NRC Safety Evaluation Reports[1]	☐	☒		
If "YES", perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.				

LBDs controlled under other regulations	YES	NO	N/A	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒		
Emergency Plan[2]	☐	☒		
Security Plan[2,3]	☐	☒		
Fire Protection Program[4] (includes the Fire Hazards Analysis)	☐	☒	☐	
If "YES", evaluate/process any changes in accordance with the appropriate regulation.				

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.5.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed.
[3] The Security Plan is classified as safeguards and can only be reviewed by personnel with the appropriate security clearance. The Preparer should notify the security department of potential changes to the Security Plan.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

(Provide a clear, concise basis for the answers given in the applicable sections above. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.)

Entergy FulFind, Electronic Text Search was utilized to search all Licensing Basis Documents (LBD's) using the keywords listed below. The only LBD impact requires modification of flow diagrams G-167 Sh2 and Sh4 referenced in FSAR Figure 6.3-1 Sh.2 and Sh4 respectively.

C. References

[Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101.]

LBDs/Documents Reviewed:
FSAR
Section 3.9.3 – ASME Code Class 1, 2 & 3 Components
Section 6.3 - Emergency Core Cooling System
Section 7.3 - Engineered Safety Features System
Section 9.3.6 – Shutdown Cooling System
Table 6.2-32 – Containment Penetration and Isolation valves

Technical Requirements Manual Table 3.6-2 – Containment Isolation Valves

Technical Specification
Section 3/4.4 – Reactor Coolant System
Section 3/4.5 – Emergency Core Cooling Sys.
Section 3/4.9.8 – Shutdown Cooling System

NUREG 800 – Standard Review Plan

NUREG 0787 – Safety Evaluation Report

Keywords:

Searched utilizing the LBDS_50_59 Table in LRS with the keywords:

SI-142, SI-1421, SI-1401, "Safety Injection bypass"

D. Is the validity of this Review dependent on any other change? *(See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)*

☐ Yes
☒ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review.

Will the proposed Change being evaluated:

Yes	No	
☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
☐	☒	Involve dredging activities in a lake, river, pond, or stream?
☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
☐	☒	Discharge any chemicals new or different from that previously discharged?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
☐	☒	Involve the installation or modification of a stationary or mobile tank?
☐	☒	Involve the use or storage of oils or chemicals?
☐	☒	Involve burial or placement of any solid wastes in the site area that may effect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

V. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

Safety Injection System piping penetrating the Reactor Containment Building at Penetrations 37 and 39 is provided with inside containment isolation check valves SI-142A and SI-142B. Each of these valves is provided with 1" diameter bypass piping and isolation valves that were provided to allow for periodic drainage of the piping downstream of the check valves. The bypass isolation valves are a potential leak path which allows nitrogen saturated water from the Safety Injection Tanks to accumulate upstream of the check valves where it creates voids in the piping. To eliminate this leak path, ER-W3-2002-0335 will remove the bypass piping and replace it with individually valved and capped drain connections.

Reason for proposed Change:

CR-WF3-2002-0818 has identified voiding in the Safety Injection piping at the containment building penetrations. Seat leakage at bypass valves SI-1421A and SI-1421B could allow nitrogen saturated water from the Safety Injection Tanks to enter the lower pressure piping upstream of Containment Isolation check valves SI-142A and SI-142B where the nitrogen comes out of solution and forms gas pockets or voids in the Low Pressure Safety Injection (LPSI) system.

50.59 Evaluation summary and conclusions

This evaluation included a review of the Licensing Basis Documents identified in Section II-A Screening, and also an Entergy-FulFind keyword search utilizing the words identified in Section II-C. The evaluation determined that the frequency of an accident or Structure, System or Component (SSC) malfunction previously evaluated in the Safety Analysis Report (SAR) will not be increased. The affected portions of the Safety Injection system are not accident initiators. The new design is consistent with the design requirements of the existing system. The proposed change will not create any new system interfaces, and will not place increased reliance on any SSC. Therefore, the proposed change does not increase the consequences of an accident or SSC malfunction, and the change does not create the possibility of an accident or SSC malfunction of a different type than previously evaluated in the SAR. The proposed change does not impact a design basis limit for a fission product barrier, and does not result in any new design basis analysis methods. There is no change to any Technical Specification, and prior approval by the Nuclear Regulatory Commission (NRC) is not required for this change. A Final Safety Analysis Report (FSAR) figure change (6.3-1 sh.4) and the Safety Injection System Failure Modes and Effects Analysis (Table 6.3-1) require revision to reflect the new drain and test connection arrangement as shown on the Document Revision Notice 02-1896 listed in Section II-A.

B. **License Amendment Determination**

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 FSAR?

☐ Yes
☒ No

BASIS:

FSAR Section 15, "Accident Analysis" identifies accidents previously evaluated in the FSAR. This chapter was reviewed and determined that the frequency of occurrence of a "Decrease in Reactor Coolant System Inventory" as described in Section 15.6.3.3, Loss Of Coolant Accident (LOCA) is the only accident which could potentially be impacted by the proposed changes. The potential for LOCA was reviewed because the new drain valves SI-1421A (B) and test connection valves SI-1401A (B) are located on a portion of the Safety Injection (SI) system that is classified as Reactor Coolant System Pressure Boundary in accordance with ANSI N18.2a-1975, which indicates that the RCS extends to the outermost containment isolation valve.

This evaluation has determined that the frequency of a LOCA will not be increased by the proposed deletion of the bypass line around safety class 1 check valve SI-142A or B because the new replacement drain (class 1) and test connection assembly (class 2), materials and design are in accordance with the applicable requirements of ASME Section III. Since the new piping configurations proposed are smaller in pipe size (3/4" vs 1" diameter); shorter in overall pipe length; and use the same branch connections off the 8" diameter header as the bypass line, the existing small break LOCA analyses encompass any postulated break of the new piping. Therefore the frequency of occurrence of an accident (LOCA) previously evaluated in the SAR is not impacted by the proposed change.

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 FSAR? ☐ Yes ☒ No

BASIS:

The proposed change will not increase the likelihood of occurrence of a malfunction of a SCC important to safety. The safety related portions of the modification are designed in accordance with the applicable requirements of ASME Section III, Class 1 and 2, and will be tested in accordance with ASME Section XI, as applicable. The proposed arrangement will be seismically and environmentally qualified for the intended application inside the containment building. The new drain and test connection valves are administratively controlled normally closed by the Operations Department and serve no active safety function other than maintaining system pressure boundary. The change does not affect any system compliance with single failure criterion.

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

BASIS:

The FSAR Chapter 15, Accident Analysis were reviewed, with the only accident consequences (radiation dose) that the proposed change could possibly impact is associated with a Loss of Coolant Accident (LOCA) as described in FSAR Subsection 15.6.3.3. The proposed modification to delete the Safety Injection bypass around SI-142A & B and replace with an independent drain and test connections will affect the previously approved LOCA analysis in the conservative direction due to the reduction of pipe size from 1" to 3/4". The existing bypass valve function is not credited as a FSAR safety function other than pressure boundary. The new drain and test connection assemblies will also serve no accident mitigating function other than pressure boundary and was considered in the design requirements material specification and procurement. The arrangement proposed by this ER will therefore not increase the consequences of a LOCA, the only previously evaluated accident analysis applicable to the subject change.

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 FSAR? ☐ Yes ☒ No

BASIS:

The proposed modification will not increase the consequences (radiological release) of any malfunction of equipment important to safety because the drain and test connection proposed will replace components of identical function presently in the Safety Injection system. The subject modification will modify the existing 1" diameter drain piping which bypasses Containment Isolation Check valve SI-142A & B by tapping off and tying back in to the 8" pipe header directly upstream and downstream of this valve.

The new arrangement proposed will use the existing pipe tap located directly downstream of the check valve for a local drain (Safety Class 1), therefore fulfilling the drain function provided by the bypass piping. The pipe tap upstream of the check valve will be utilized for a test connection (Safety Class 2), matching the present system design which utilizes a connection off the bypass line as it ties into the header upstream of the check valve. Since the design functions of the Safety Injection system are met by the new proposed arrangement, without creation of any new system interactions no increase in consequences of a SSC malfunction can be realized.

The implementation of the proposed change will also not create the possibility of increased consequences due to a malfunction of the shutdown cooling system or containment integrity because the work will either be performed when the core is offloaded, or when Technical Specification requirements pertaining to Shutdown Cooling and Containment Building integrity are met.

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

BASIS:

The similarities between the new drain and test connection arrangement and the existing bypass with test connection do not create any system interactions or new accident conditions that are not encompassed by the existing system design basis and previously analyzed accident scenarios. Replacement of the drain bypass piping around valve SI-142A & B with an new independent local drain and test connection meets the previously approved system functions without the creation of any new accident types or scenarios.

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 FSAR? ☐ Yes ☒ No

BASIS

The proposed modification cannot create the possibility for a malfunction of a SSC important to safety with a different result than any previously evaluated in the FSAR because the existing safety related piping and valves will be replaced with components that meet the same ASME Section III Class 1 and 2 requirements. The Safety Injection design basis will not be impacted by the proposed modification that replaces the functions of system draining and valve testing using assemblies consisting of qualified safety class code components and materials. The new independent drain and test connection configuration will also be analyzed for seismic and environmental qualification considerations to ensure acceptability for use in the subject applications. The change could not lead to a failure with a different result than previously evaluated in the FSAR Failure Modes and Effects Analysis.

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

BASIS:

The purpose of the proposed configuration change is to restore or maintain the fission product barrier provided by the Reactor Coolant System (RCS) pressure boundary and the Safety Injection (SI) system by eliminating a leak path from the Safety Injection Tanks. The new proposed independent drain and test connection are qualified replacements for the existing bypass line around the inside containment isolation valve SI-142A & B. The location is on a portion of the SI system that is classified as part of the RCS pressure boundary because the RCS boundary extends to the outermost containment isolation valve as described in ANSI N18.2a-1975. However, existing ASME Section III, Class 1 check valves SI-335B (downstream of SI-142B), and SI-336B (downstream of SI-142A) provide the first safety related barrier to prevent backflow of RCS inventory. The new Safety Class 1 drain valves SI-1421A & B in conjunction with check valves SI-142A & B provides the second barrier prior to the system Safety Class 2 classification for the containment penetration piping which includes the new SI-1401A & B test connection. Therefore the integrity of the Reactor Coolant System, Safety Injection System and Containment Isolation System will not be degraded by the proposed changes.

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

BASIS:

The proposed changes to install a local independent drain and test connection will provide the same design features (test and drain), presently provided by the bypass line on the Safety Injection System and does not require re-analysis of any existing design basis or safety analysis; therefore it does not affect any existing method of evaluation or analytical method described in the FSAR. The design basis or method of evaluating the design basis for the Safety Injection system, and containment penetration, are described in W3-DBD-001 "Safety Injection", and W3-DBD-026 "Containment Isolation and Leakage Rate Testing" are unaffected.

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 Steam Electric Station

Document Reviewed: ER-W3-2002-0585-00_____ **Change/Rev.:** N/A

System Designator(s)/Description: CCW

Description of Proposed Change

Background: Valves CC-641, CC-710 and CC-713 are ten inch diameter, air operated (fail open, air to close) containment isolation valves for a nonessential portion (Reactor Coolant Pumps (RCP) and Control Element Drive Mechanism (CEDM) cooling coils) of the Component Cooling Water (CCW) System. Valve CC-641 is the upstream outside containment isolation valve and valves CC-710 and CC-713 are the downstream inside and outside containment isolation valves respectively. DC-3493 installed flow control valves (CC-6411, CC-7101 and CC-7131) in the air supply lines to the CC-641, CC-710 and CC-713 operators to increase the closing times to reduce hydraulic transients that were resulting in pressures higher than the system design pressure in the Component Cooling Water (CCW) System. The acceptance testing for DC-3493 per Special Test Procedure (STP) 99003493 revealed that the pressure transients continue to occur with peak pressures still exceeding the system design pressure. The Design Basis maximum (slowest) closure time for valves CC-641, CC-710 and CC-713 is currently five seconds maximum. The five second maximum closure is also in the Waterford 3 Updated Final Safety Analysis Report (UFSAR) and Technical Requirements Manual (TRM). In addition, no design basis minimum (fastest) closure times have been established for valves CC-641, CC-710 and CC-713.

The proposed change consists of the following:

> Implement minimum and maximum close times for valves CC-641, CC-710 and CC-713 such that the minimum closing time results in a peak pressure in the CCW system piping that is within the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code allowable operating pressure, and the maximum closing time remains below the regulatory/industry standard maximum closing time. The design basis minimum closure time has been determined to be five seconds. Test data from STP 99003493 was used in the determination of the minimum closure time. The design basis maximum closure time has been established at fifty seconds. The maximum closure time of fifty seconds is in accordance with American National Standards Institute (ANSI) standard ANSI-N271-1976, "Containment Isolation Provisions for Fluid Systems" which is endorsed by Nuclear Regulatory Commission (NRC) Regulatory Guide 1.141.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____.

(Insert item # from Section 5.2.2.2).

☐ The proposed activity is controlled by another regulation per Section 5.2.2.3.

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**SCREENING**	**Sections I, II, and III required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, III, and IV required**
☒	**50.59 EVALUATION (#: ____)** 03-006	**Sections I, II, III, and V required**

Preparer: R.T. Finch / R.T. Finch EOI / Design Engineering / March 27, 2003
Name (print) / Signature / Company / Department / Date

Reviewer: J. Russo / EOI / Design Engineering / 3/31/03
Name (print) / Signature / Company / Department / Date

OSRC Jerry A. Ridgel / 4/9/03
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	**Scope of Assistance:**
None.	N/A

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents? (Check "N/A" for those documents that are not applicable to the facility.)**

Operating License	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒		
TS	☐	☒		
NRC Orders	☐	☒	☐	
If "YES", obtain NRC approval prior to implementing the change. (See Section 5.1.13 for exceptions.)				

LBDs controlled under 50.59	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☒	☐		DRN 03-169 (UFSAR Chapter 6)
TS Bases	☐	☒		
Technical Requirements Manual	☒	☐		DRN 03-168 (Table 3.3-5)
Core Operating Limits Report	☐	☒		
Offsite Dose Calculations Manual	☐	☒		
NRC Safety Evaluation Reports[1]	☐	☒		
If "YES", perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.				

LBDs controlled under other regulations	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒		
Emergency Plan[2]	☐	☒		
Security Plan[2, 3]	☐	☒		
Fire Protection Program[4] (includes the Fire Hazards Analysis)	☐	☒	☐	
If "YES", evaluate/process any changes in accordance with the appropriate regulation.				

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.5.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed.
[3] The Security Plan is classified as safeguards and can only be reviewed by personnel with the appropriate security clearance. The Preparer should notify the security department of potential changes to the Security Plan.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. **Basis**

(Provide a clear, concise basis for the answers given in the applicable sections above. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.)

UFSAR Table 6.2-32 currently lists the maximum closure time for valves CC-641, CC-710 and CC-713 as five seconds. Table 3.3-5 of the TRM currently lists the maximum closure time for the "CCW to RCP Valves" as less than or equal to twelve seconds with offsite power available and less than or equal to twenty-three and one-half seconds for loss of offsite power. The new design basis closure times for CC-641, CC-710 and CC-713 are outside of the ranges currently in the UFSAR and TRM. This will require changes to the UFSAR and TRM to reflect the new design basis minimum and maximum close times.

A separate 50.59 review will address testing (via a Special Test Procedure) for the proposed change. For this reason Question No. 2 above, concerning testing, was answered No.

Waterford 3 does not have an Independent Spent Fuel Storage Installation so question No. 3 above was answered N/A.

C. References

[Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101.]

Entergy FullFind and Licensing Research System (LRS) software was utilized to search the LBD's associated with Waterford 3. LBDs that were reviewed include the UFSAR, UFSAR Questions, the Technical Specifications, Commitments, Correspondence, the Technical Requirements Manual and LBDS_50_59. The keywords utilized were as follows: "component cooling water"; "pressure surge"; "hydraulic transient"; "CC-710"; "CC-713"; "containment spray actuation"; "containment isolation". Manual searches of the Waterford 3 GSB Library Controlled UFSAR and TRM were also performed for the sections where electronic hits occurred. The Waterford 3 Operating Experience and Institute of Nuclear Power Operations (INPO) databases were also searched using the key words "water hammer" and "CCW water hammer".

Documents:

1. Waterford SES Unit No. 3 Component Cooling Water Auxiliary Component Cooling Water Design Basis Document (W3-DBD-004).
2. Waterford SES Unit No. 3 Safety Related, Air Operated Valves Design Basis Document (W3-DBD-014).
3. U.S. Nuclear Regulatory Commission (N.R.C.) Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station Unit No. 3, July 1981 – Section 6.2.4, Containment Isolation System
4. U.S. NRC Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems".
5. American National Standard ANSI-N271-1976, "Containment Isolation Provisions for Fluid Systems".
6. U.S. NRC Generic Letter No. 83-10a, Resolution of TMI Action Item II.K.3.5, Automatic Trip of RCPs

FSAR Sections Reviewed:

1. UFSAR Section 1.9.28 Containment Isolation Dependability.
2. UFSAR Section 3.1.40, Criterion 44 – Cooling Water.
3. UFSAR Section 3.1.49, Criterion 56 – Primary Containment Isolation
4. UFSAR Section 5.4.1, Reactor Coolant Pumps
5. UFSAR Section 6.2.4, Containment Isolation System.
6. UFSAR Section 9.2.2, Cooling System for Reactor Auxiliaries

FSAR Sections Reviewed Continued:

7. UFSAR Section 15, Accident Analyses
8. UFSAR Appendix 15B, Containment Leakage and Dose Calculation Models

FSAR Tables Reviewed:

1. UFSAR Table 1.9-3, "Containment Isolation Valves Provided with Capability for Manual Operation".
2. UFSAR Table 6.2-32, Containment Penetrations and Isolation Valves

FSAR Questions Reviewed:

1. Question No. 480.44

Technical Requirements Manual Sections Reviewed:

1. Table 3.3-5 "Engineered Safety Features Response Times"

Technical Specifications Reviewed:

1. 3/4.3.2, Engineered Safety Features Actuation System Instrumentation.
2. 3/4.6.3, Containment Isolation Valves.
3. 3/4.7.3, Component Cooling Water and Auxiliary Component Cooling Water Systems
4. 4.05, Surveillance Requirements for Inservice Inspection and testing of ASME Code Class 1, 2 and 3 components.

INPO Events: Nos. 285-961111-1, 261-920618-1, 382-960228-1 and 346-981014-1

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☐ Yes
☒ No

If "Yes," list the required changes.

N/A

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review.

Will the proposed Change being evaluated:

Yes	No	
☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
☐	☒	Involve dredging activities in a lake, river, pond, or stream?
☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
☐	☒	Discharge any chemicals new or different from that previously discharged?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
☐	☒	Involve the installation or modification of a stationary or mobile tank?
☐	☒	Involve the use or storage of oils or chemicals?
☐	☒	Involve burial or placement of any solid wastes in the site area that may effect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

V. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after PSRC approval, if available.)

Brief description of change, test, or experiment:

The proposed change consists of the following:

Implement minimum and maximum close times for valves CC-641, CC-710 and CC-713 such that the minimum closing time results in a peak pressure in the CCW system piping that is within the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code allowable operating pressure and the maximum closing time remains below the regulatory/industry standard (Regulatory Guide 1.141 and ANSI-N271-1976) maximum closing time. The design basis minimum closure time has been determined to be five seconds and the design basis maximum closure time has been established at fifty seconds. The minimum and maximum closure times are in accordance with the requirements / standards discussed above.

The proposed change will require changes to the Updated Final Safety Analysis Report Chapter 6, Technical Requirements Manual Table 3.3-5, OP-002-003 and OP-903-036.

Reason for proposed Change:

The design pressure for this portion of the Component Cooling Water (CCW) system is 125 psig and the peak momentary (less than one second duration) pressure that is calculated to result from the current two second CSAS closure of CC-713 concurrent with the postulated single active failure of CC-641 (failure to close on CSAS) is 278 psig.

50.59 Evaluation summary and conclusions

The proposed change to the CCW system between and including valves CC-641 and CC-713 will not impact the ability of the system to perform its design function. Resetting the closing times of valves CC-641, CC-710 and CC-713 to a minimum of 5 seconds and a maximum of 50 seconds will ensure that the pressures resulting from the minimum closing time are below the ASME allowable operating pressure and the maximum closing times are within the maximum time established in Regulatory Guide 1.141 and ANSI-N271-1976. There will be no impact on the remainder of the CCW system or on any other plant systems, structures or components. There is no impact on containment integrity or the ability to isolate containment. There are no accidents evaluated in the UFSAR caused by the CCW system. The proposed change does not impact any controlling numerical values established in the licensing basis for a fission product barrier. The proposed change does not increase the potential for a malfunction of a structure, system or component (SSC) and does not place more reliance on a SSC for

accident mitigation. The proposed change involves no new evaluation methods (as described in the UFSAR). All evaluation questions were answered "No".

B. License Amendment Determination

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 FSAR? ☐ Yes ☒ No

BASIS:

The affected portion of the CCW system is between and includes valves CC-641 and CC-713. Valves CC-641, CC-710 and CC-713 are ten inch diameter, air operated (fail open, air to close) containment isolation valves for a nonessential portion (Reactor Coolant Pumps (RCP) and Control Element Drive Mechanism (CEDM) cooling coils) of the CCW System. Valve CC-641 is the upstream outside containment isolation valve for containment penetration No. 23 and valves CC-710 and CC-713 are the downstream inside and outside containment isolation valves respectively for containment penetration No. 24.

The proposed change consists of the following:

> Implement minimum and maximum close times for valves CC-641, CC-710 and CC-713 such that the minimum closing time results in a peak pressure in the CCW system piping that is within the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code allowable operating pressure and the maximum closing time remains below the regulatory/industry standard maximum closing time.

The UFSAR was reviewed and there are no UFSAR accidents that are caused by the Component Cooling Water (CCW) System. The proposed change will therefore not increase 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 FSAR? ☐ Yes ☒ No

BASIS:

Valves CC-641, CC-710 and CC-713 are ten inch diameter, air operated (fail open, air to close) containment isolation valves for a nonessential portion (RCP and CEDM cooling coils) of the CCW System. These valves close upon receipt of a Containment Spray Actuation Signal. As containment isolation valves they are components important to safety previously evaluated in the UFSAR. These valves are listed in UFSAR Table 6.2.32, "Containment Penetrations and Isolation Valves". Valve CC-641 is the outside containment isolation valve for containment penetration No. 23 and valves CC-710 and CC-713 are the inside and outside containment penetration valves for penetration No. 24 respectively. These valves,

and penetrations, are categorized as General Design Criterion (GDC) 56, Primary Containment Isolation, valves and penetrations. The applicable design requirement from GDC 56 is as follows:

> One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Note that valve CC-644 is the inside containment isolation valve for containment penetration No. 23 and this valve is a check valve.

The other provisions of GDC 56 are not applicable since CCW flow must be provided to the RCPs and CEDM Coolers during plant operation (i.e. it would not be possible to have a locked closed isolation valve in this portion of the CCW system since flow must be provided to the RCP and CEDM coolers during plant operation).

Valves CC-641, CC-710 and CC-713 were exempted from the requirement to close on a Safety Injection Actuation Signal (SIAS) or Containment Isolation Actuation Signel (CIAS) and have the capability to be reopened via manual override because it was determined that the "safe" post accident condition for these valves could be either open or closed, depending on the post accident conditions inside containment. The open position would be desirable if it were decided to restore cooling water flow to the Reactor Coolant Pumps and Control Element Drive Mechanism Coolers. These valves were therefore exempted from certain provisions of Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems" in accordance with the intent of the guidance contained in Generic Letter 83-10a (Automatic Trip of Reactor Coolant Pumps). These valves are also designed to fail "open" on loss of power or air because open is the position of greater safety for these valves (to protect the Reactor Coolant Pumps and Control Element Drive Mechanisms).

The proposed change (see the answer to question No. 1 above) will ensure that the piping and components in the affected portion of the CCW system will remain within the design basis allowable stresses and will function in accordance with the functional requirements assumed in the licensing basis safety analysis. The redundancy of having two isolation valves (inside and outside containment) at each containment penetration is also not impacted by the proposed change.

The design basis minimum closure time has been determined to be five seconds. Test data from STP 99003493 was used in the determination. The minimum design basis closure time of 5 seconds for valves CC-641, CC-710 and CC-713 will ensure that the operating pressures in the affected portions of the CCW system are less than the ASME Code allowable operating pressure.

The design basis maximum closure time has been established at fifty seconds. The maximum design basis closure time has been established in accordance with American National Standard ANSI-N271-1976 (Containment Isolation Provisions

for Fluid Systems). This standard is endorsed by the U.S. Nuclear Regulatory Commission in Regulatory Guide 1.141 (Containment Isolation Provisions for Fluid Systems). Section 4.4.4 (Valve Closure Time) of ANSI-N271-1976 states that "In general, power-operated valves 3 ½ inches to 12 inches in diameter should be closed at least within a time determined by dividing the nominal valve diameter by 12 inches per minute. (As an example, a 12-inch diameter valve would be required to close within one minute, a six-inch valve within 30 seconds)." Using this guidance the maximum closure time for the 10 inch diameter CCW valves would be 10/12 minutes or 50 seconds. The ANSI standard also states that "In determining valve closure time, consideration must be given to fluid system requirements (e.g. water hammer) and the effect of closure time on valve reliability, as well as containment isolation requirements".

Systems that connect directly from the containment atmosphere to outside containment have a regulatory requirement to close in a short time after the initiation of an accident. FSAR Section 6.2.4.2.3 states that "the closure time for the Containment Purge System isolation valves have been chosen at 5 seconds to limit radiological effects after the fuel handling accident within the guide lines established by 10CFR100. The closure times of the remaining isolation valves have been conservatively selected as shown in Table 6.2-32". FSAR Table 6.2-32 currently lists a five second maximum closure time for the CC-641, CC-710 and CC-713 valves but these valves are not Containment Purge System valves and so the 5 second maximum close time is not a regulatory requirement for these valves.

The Nuclear Regulatory Commission (NRC) Safety Evaluation Report for Waterford 3 states in Section 6.2.4 (Containment Isolation System) that "Valve closure will occur within 60 sec. with most valves closing in 10 sec. or less. The containment purge system isolation valves are designed to close in 5 sec. These valve closure times are acceptable".

The NRC Standard Review Plan Section 6.2.4 states that "In general, valve closure times should be less than one minute, regardless of valve size. Valves that provide a direct path to the environs, e.g., the containment purge and ventilation system lines and main steam lines for direct cycle plants, may have to close in times much shorter than one minute". Valves CC-641, CC-710 and CC-713 do not provide a direct path between the containment atmosphere and the outside atmosphere.

Increasing the closing time of valves CC-641, CC-710 and CC-713 will require that valves CC-6411, CC-7101 and CC-7131 respectively be adjusted in the closed direction relative to their present positions. This will not introduce the possibility of malfunction due to clogging of the air supply line/valve orifice because the instrument air system provides a supply of dry, oil free, filtered air.

In summary:

1. The new minimum and maximum closure times for valves CC-641, CC-710 and CC-713 are within the regulatory/industry standard (U.S. NRC Regulatory Guide 1.141 and ANSI-N271-1976) close time requirements for containment isolation valves.
2. The proposed change adds no new components and does not impact the function, redundancy or reliability of the affected plant components

The likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR therefore remains unchanged.

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

BASIS:

For this question "consequences of an accident "refers to the radiation dose associated with the plant's response to an accident or malfunction.

The proposed change does not change the function or ability to perform the function of the affected components of the CCW system. No new components are added by the proposed change.

Valves CC-641, CC-710 and CC-713 are ten inch diameter, air operated (fail open, air to close) containment isolation valves for a nonessential portion (RCP and CEDM cooling coils) of the CCW System. As containment isolation valves they are components that could result in additional release of radioactivity if they failed to perform as designed (to close on a CSAS). These valves are listed in UFSAR Table 6.2.32, "Containment Penetrations and Isolation Valves".

The only physical change being made is to slow the close stroke time of CC-641, CC-710 and CC-713 from 5 seconds maximum to 50 seconds maximum and to establish a new minimum stroke closed time of 5 seconds. This has no impact on the function of these valves as given in the UFSAR safety analyses to close on a CSAS to provide containment isolation. The increase in close time for the valves also has no impact on any potential radioactive release because there are no existing dose calculations that specifically include penetrations No. 23 and 24 as a radioactive release path. In any event the increase in dose would be insignificant and would be bounded by the letdown line break (UFSAR Section 15.6.3). There is thus no increase in dose to the public due to the proposed change.

There is thus no possibility of increasing 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 FSAR? ☐ Yes ☒ No

BASIS:

The proposed change is to slow the close stroke time of valves CC-641, CC-710 and CC-713 from 5 seconds maximum to 50 seconds maximum and to establish a new minimum stroke closed time of 5 seconds. There are redundant containment isolation valves for each penetration and as described above the proposed change does not change the function or ability to perform the function of the affected valves. The consequences of a malfunction of any of the affected containment isolation valves after the proposed change is implemented will be identical to the consequences prior to the proposed change. The proposed change also does not require that greater reliance be placed on any system, structure or component.

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

BASIS:

The proposed change does not create the possibility of an accident of a different type than any previously evaluated in the UFSAR. No new components are installed by the proposed change and no new system interactions are being created. The affected parts of the CCW system will operate as they did previously with the exception that valves CC-641, CC-710 and CC-713 will close slower on a CSAS. Instead of closing in a maximum time of 5 seconds the valves will now have a maximum close time of 50 seconds.

Waterford 3 Operating Procedure OP-901-504, "Inadvertent ESFAS Actuation" requires that CCW flow to the RCPs be restored within three minutes following an inadvertent CSAS to prevent damage to the RCP seals. The CC-641, CC-710 and CC-713 valves are spring to open and air to close. Per drawing B430 Sheet V24 the air exhaust vent, which opens to depressurize the actuator to allow the spring to open the valve, is downstream of the air metering valves (CC-6411, CC-7101 and CC-7131). Since the location and functioning of the solenoid is not impacted by the proposed change there will be no change to the opening time of valves CC-641, CC-710 and CC-713. There will thus be no impact on the ability to open the valves following an inadvertent closure.

A new accident of a different type than any previously evaluated in the UFSAR can therefore not be created by the proposed change.

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

BASIS

As stated above the only change being made is to slow the close stroke time of CC-641, CC-710 and CC-713 from 5 seconds maximum to 50 seconds maximum and to establish a new minimum stroke closed time of 5 seconds. All of the components affected by the proposed change will function in exactly the same manner as prior to the change with the exception that the valves will close slower on a CSAS. The proposed change does not remove or replace any of the affected components or install any new components.

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

BASIS:

As described in the answer to Question No. 3 above the proposed change has no impact on the ability of valves CC-641, CC-710 and CC-713 to perform their safety function of containment isolation. Resetting the closing times of valves CC-641, CC-710 and CC-713 to a minimum of 5 seconds and a maximum of 50 seconds will ensure that the pressures resulting from the minimum closing time are below the ASME allowable operating pressure and the maximum closing times are within the maximum time established in Regulatory Guide 1.141 and ANSI-N271-1976. Although the closing stroke time of containment isolation valves CC-641, CC-710 and CC-713 is being impacted by the proposed change, the change does not impact the UFSAR accident analyses or a design basis barrier limit.

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

BASIS:

The proposed change involves no new method of evaluation. The methods used to determine the acceptability of the new design basis closing times for valves CC-641, CC-710 and CC-713 use the same NRC Regulatory Guides and ANSI Standards as used for the original plant design and licensing.

I. OVERVIEW / SIGNATURES

Facility: Waterford 3

Document Reviewed: ER-W3-2003-0257-000 **Change/Rev.** _____

System Designator(s)/Description: Containment Isolation System

Description of Proposed Change

This change revises the TS Bases associated with TS 3.6.3, Containment Isolation Valves, Note 8 of TRM 3.6-2 and the TRM Bases associated with TRM 3.6.3. Specifically, the statement "Locked or sealed closed valves may be open on an intermittent basis under administrative control" as contained in each of the above documents will be changed. This change will clarify that a deactivated automatic valve secured in the isolation position is equivalent to a locked or sealed closed valve and that this statement is applicable to a deactivated automatic valve secured in the isolation position.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert Item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☑	50.59 EVALUATION (#: 03-007)	Sections I, II, III, IV, and VI required

Preparer: Michael K Brandon/ [signature] /EOI/Licensing/ May 27, 2003
Name (print) / Signature / Company / Department / Date

Reviewer: Lisa B. Borel/ Lisa B. Borel /EOI/Licensing/ may 27, 2003
Name (print) / Signature / Company / Department / Date

OSRC: Jerry A. Ridgel for Ken Peters per telecon [signature] 5/29/03
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:

II. SCREENING

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☑	
TS	☐	☑	
NRC Orders	☐	☑	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☐	☑	
TS Bases	☑	☐	DRN 03-666 TS Bases Page B3/4 6-5 (Section 3/4.6.3)
Technical Requirements Manual	☑	☐	DRN-03-667 TRM Pages 3/4 6-17 and B 3/4 3a (Sections – Table 3.6-2 and 3/4 .6.3, respectively)
Core Operating Limits Report	☐	☑	
NRC Safety Evaluation Reports[1]	☐	☑	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☑	
Emergency Plan[2]	☐	☑	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☑	
Offsite Dose Calculations Manual	☐	☑	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ Yes ☑ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation? ☐ Yes ☐ No ☑ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. **Basis**

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

This change is intended to clarify that it is acceptable to open a deactivated automatic valve secured in its isolation position on an intermittent bases under administrative controls. This change will document in the TS Bases and the TRM and its basis that the asterisk footnote annotated to the Waterford TS 3.6.3 Limiting Condition for Operation applies to deactivated automatic valves as well as to locked or sealed closed valves. Although not explicitly stated in the current Waterford TS Bases for TS 3.6.3, a deactivated automatic valve secured in the isolated position is functionally equivalent to a locked or sealed close valve. It should be noted that the ANO Unit 2 TS 3.6.3 is identical to the Waterford 3 TS 3.6.3 with respect to this footnote and the ANO TS 3.6.3 Bases explicitly allows the use of administrative controls for deactivated automatic valves. The words in the ANO Bases were reviewed and approved by the NRC as part of the Licensing Amendment submitted by ANO which added the subject footnote (References 1 and 2). It should also be noted that the CE Improved TS, NUREG 1432 Revision 2, (Reference 3) has a similar note and it is applied in a broader sense such that a deactivated automatic valve is clearly within the scope of the note. The note in NUREG 1432 allows the unisolation of penetration flow paths under administrative control. The NUREG 1432 does not limit the types of valves this note is applied to. Based on this information, the changes proposed do not impact the Waterford 3 TS and is consistent with the current intent of the TS.

The use of administrative controls for the intermittent opening of closed containment isolation valves is not discussed in the FSAR and this activity does not represent a new test or experiment.

C. **References**

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
A fulfind electronic search of the 50.59 LBDs was performed. Discussions of deactiviated or lock and sealed valves were identified in TS sections 3.4, 3.6 and 3.7. Additionally a search using the keyword of "intermittent" was performed on the FSAR.	"deactivated automatic valve", "locked or sealed closed" and "intermittent"

LBDs/Documents reviewed manually:

Reference 1: The SER on ANO Unit 2 License Amendment 154.

Reference 2: ANO Letter 2CAN079302 dated 7/22/93 RE: TSCR to Relocate Containment Isolation Valve Table.

Reference 3: NUREG 1432, Rev, 2, "Standard Technical Specifications Combustion Engineering Plants."

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
☑ No

If "Yes," list the required changes.

_____

_____

_____

_____

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☑	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☑	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☑	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☑	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☑	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☑	Discharge any chemicals new or different from that previously discharged?
7.	☐	☑	Change the design or operation of the intake or discharge structures?
8.	☐	☑	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☑	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☑	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☑	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☑	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☑	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☑	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☑	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☑	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☑	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☑	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☑	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☑	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☑	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☑	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☑	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☑	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☑	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) _____ ☐ No

_____

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

This change revises the TS Bases associated with TS 3.6.3, Containment Isolation Valves, Note 8 of TRM Table 3.6-2 and the TRM Bases associated with TRM 3.6.3. Specifically, the statement "Locked or sealed closed valves may be open on an intermittent basis under administrative control" as contained in each of the above documents will be changed. This change will revise this statement to acknowledge that deactivated automatic containment isolation valves secured in the isolation position may be open on an intermittent basis under administrative control.

Reasons for proposed Change:

This change is intended to clarify the scope of the TS 3.6.3 LCO footnote that reads, "Locked or sealed closed valves may be opened on an intermittent basis under administrative control." The review of the current Standard Technical Specifications for Combustion Engineering Plants (NUREG 1432 Rev. 2) and the ANO Unit 2 TS verify that this clarification is appropriate and consistent with the intent of this statement. This TS Basis change is being made to ensure that the TS requirements are used correctly and consistently by the various Operations Shift crews. Revising the TS Basis as a means for documenting this clarification is consistent with the guidance provided by NRC in Information Notice 97-80.

50.59 Evaluation summary and conclusions

This change revises the TS Bases associated with TS 3.6.3, Containment Isolation Valves, Note 8 of TRM Table 3.6-2 and the TRM Bases associated with TRM 3.6.3. This change is administrative in nature and does not change any requirement or the intent of the associated TS requirement. This change addresses the use of administrative controls associated with closed containment isolation valves. Containment isolation valves are not initiators of any accident evaluated in the FSAR. Therefore this change will not result in any increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

This change will clarify that the use of administrative controls for the intermittent opening of lock closed or sealed closed containment isolation valves is also applicable to closed and deactivated automatic valves. All of these three types of valve configurations constitutes passive containment isolation barriers and are therefore functionally similar. The current Licensing Basis for Waterford, as specified in the Waterford Technical Specifications, allows the use of administrative controls for the intermittent opening of passive administratively controlled containment isolation barriers. While one could postulate a human error could affect the successful closure of a containment isolation valve that was opened under administrative controls, this change does not represent any increase to likelihood of this occurrence since this likelihood is currently a part of the current Licensing basis. Therefore this change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

The current Licensing Basis for Waterford, as specified in the Waterford Technical Specifications, allows the use of administrative controls for the intermittent opening of passive administratively controlled containment isolation barriers. The requirements for these administrative controls are explicitly defined in the current TS Bases and in the applicable NRC Safety Evaluation Reports. These rigorous requirements (i.e., (1) stationing an operator, who is in constant communication with the control room, at the valves controls (2) instructing this operator to close these valves in an accident situation; and (3) assuring that environmental conditions will not preclude access to close

the valves and that this action will prevent the release of radioactivity outside the containment) for these controls ensures that all containment penetrations can be isolated to limit radioactive releases. Therefore this change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

This change does not represent any increase to consequences of a malfunction since a malfunction of this type has been considered by the NRC as part of the Licensing of the plant and determined to not be credible based on the rigorous requirements stipulated for the administrative controls. Therefore this 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 FSAR.

This change does not affect or create an initiator of any accident. Therefore this change will not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

This change does not introduce physical changes to any existing equipment in the plant, does not introduce any new type of equipment in the plant, and does not introduce any change in the administrative requirements for ensuring safe operation of the plant. Therefore 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 FSAR.

This change will not affect the design basis limit of this fission product barrier. The administrative controls involved are currently part of the plant's Licensing Basis. No other fission product barriers are associated with this change. Therefore this change will not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. In addition, this change does not involve any method of evaluation. Therefore this change does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

B. License Amendment Determination

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 FSAR? ☐ Yes ☑ No

   BASIS:

   This change revises the TS Bases associated with TS 3.6.3, Containment Isolation Valves, Note 8 of TRM Table 3.6-2 and the TRM Bases associated with TRM 3.6.3. This change is administrative in nature and does not change any requirement or the intent of the associated TS requirement. This change addresses the use of administrative controls associated with closed containment isolation valves. Containment isolation valves are not initiators of any accident evaluated in the FSAR. Therefore this change will not result in any increase in 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 FSAR? ☐ Yes ☑ No

   BASIS:

   This change will clarify that the use of administrative controls for the intermittent opening of lock closed or sealed closed containment isolation valves is also applicable to closed and deactivated automatic valves. All of these three types of valve configurations constitutes passive containment isolation barriers and are therefore functionally similar. The current Licensing Basis for Waterford, as specified in the Waterford Technical Specifications, allows the use of administrative controls for the intermittent opening of passive administratively controlled containment isolation barriers. While one could postulate a human error could affect the successful closure of a containment isolation valve that was opened under administrative controls, this change does not represent any increase to likelihood of this occurrence since this likelihood is currently a part of the current Licensing basis. The requirements for these administrative controls are rigorous and explicitly defined in the current TS Bases and in the applicable NRC Safety Evaluation Reports. These rigorous requirements (i.e., (1) stationing an operator, who is in constant communication with the control room, at the valves controls (2) instructing this operator to close these valves in an accident situation; and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment) provide an effective means for minimizing any risk associated with this manual action and provide reasonable assurance that a human error will not be a concern. Note that the grouping of deactivated automatic valves with locked or sealed valves is explicitly done in the footnote associated with TS 3/4.6.1.1. The 3/4.6.1.1 footnote states, "...valves, blind flanges, and deactivated automatic valves ...and are locked, sealed or otherwise secured in the closed position." Therefore this change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

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

BASIS:

This change will clarify that the use of administrative controls for the intermittent opening of locked closed or sealed closed containment isolation valves is also applicable to closed and deactivated automatic valves. All of these three types of valve configurations constitutes passive containment isolation barriers and are therefore functionally similar. The current Licensing Basis for Waterford, as specified in the Waterford Technical Specifications, allows the use of administrative controls for the intermittent opening of passive administratively controlled containment isolation barriers. The requirements for these administrative controls are explicitly defined in the current TS Bases and in the applicable NRC Safety Evaluation Reports. These rigorous requirements (i.e., (1) stationing an operator, who is in constant communication with the control room, at the valves controls (2) instructing this operator to close these valves in an accident situation; and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment) for these controls ensures that all containment penetrations can be isolated to limit radioactive releases. Therefore this change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

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

BASIS:

This change will clarify that the use of administrative controls for the intermittent opening of locked closed or sealed closed containment isolation valves is also applicable to closed and deactivated automatic valves. All of these three types of valve configurations constitutes passive containment isolation barriers and are therefore functionally similar. The current Licensing Basis for Waterford, as specified in the Waterford Technical Specifications, allows the use of administrative controls for the intermittent opening of passive administratively controlled containment isolation barriers. While one could postulate a human error could affect the successful closure of a containment isolation valve that was opened under administrative controls, this change does not represent any increase to consequences of a malfunction since a malfunction of this type has been considered by the NRC as part of the Licensing of the plant and determined to not be credible based on the rigorous requirements stipulated for the administrative controls. Therefore this 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 FSAR.

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

BASIS:

This change is administrative in nature and does not affect or create an initiator of any accident. Therefore this change will not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

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

BASIS

This change is administrative in nature and does not introduce physical changes to any existing equipment in the plant, does not introduce any new type of equipment in the plant, and does not introduce any change in the administrative requirements for ensuring safe operation of the plant. Therefore 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 FSAR.

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

BASIS:

While this change does involve the application of administrative controls associated with ensuring the effectiveness of the containment (a fission product barrier), this change will not affect the design basis limit of this fission product barrier. The administrative controls involved are currently part of the plant's Licensing Basis. No other fission product barriers are associated with this change. Therefore this change will not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered

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

BASIS:

This change is administrative in nature and does not involve any method of evaluation. Therefore this change does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses

I. OVERVIEW / SIGNATURES

Facility: _____Waterford 3___

Document Reviewed: _ER-W3-2003-0112-000 **Change/Rev.** _____0__

System Designator(s)/Description: _____Safety Injection

Description of Proposed Change
ER-W3-2003-0112-000 evaluates the LPSI train "B" piping with the presence of voids (up to 0.8 ft^3) in the entire LPSI B train. This condition is being incorporated into the design of the LPSI Train "B" discharge piping. The new void limit will revise the existing limit in OP-903-026. Technical Specification Basis 3/4.5.2 and 3/4.5.3 are revised to address the void, which is applicable to LPSI train "B" discharge piping only (DRN 03-0445). W3-DBD -001 has also been revised for LPSI train B by DRN 03-0446

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____.
(Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
■	50.59 EVALUATION (#: 03-008)	Sections I, II, III, IV, and VI required

Preparer: Dipak Dasgupta/ Dipak Dasgupta 5/29/03
Name (print) / Signature / Company / Department / Date

Reviewer: Nara Ray Nara Ray 5/29/03
Name (print) / Signature / Company / Department / Date

OSRC: Jerry A. Ruhl for Ken Peters on Fleet 5/29/03
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
_Dan Rohli	__Mechanical System
Michelle Groome	I&C
Nasser Pazooki	SEA

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	■	
TS	☐	■	
NRC Orders	☐	■	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☐	■	
TS Bases	■	☐	DRN 03-0445, Section 3/4.5 page B 3 / 4 5-1d
Technical Requirements Manual	☐	■	
Core Operating Limits Report	☐	■	
NRC Safety Evaluation Reports[1]	☐	■	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	■	
Emergency Plan[2]	☐	■	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	■	
Offsite Dose Calculations Manual	☐	■	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ■ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ■ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. **Basis**

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

Entergy-Fulfind, Electronic Text Search Utility using the LBDS 5059 index, was utilized to search all the LBD's.

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Yes. Technical Specification Section 3/4.5, page B 3/4 5-1d has been revised to include void in the LPSI B system. DRN 03-0445 has been generated to initiate the change. 50 59 evaluation has been performed.

W3-DBD-001 has been revised to include the void in the LPSI B train.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?**

No. There is no proposed test or experiment in this ER. The presence of voids in the discharge piping of LPSI train "B" will cause Hydraulic transient loading and a pressure spike in the system. The piping has been evaluated for this transient and the piping and supports remain within the ASME Code allowables. The pressure spike will not exceed the current design pressure of the system. The hydraulic transient loading will not have any adverse effect on the remaining portions of the system. The loading on the pump nozzles are still within the vendors design allowables. The loading on the containment penetration is still within the design allowables.

3 **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?**

Not applicable for this ER.

C. **References**

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:

LBDs/Documents Reviewed:

FSAR Sections:

1.2.2.7.4; 1.9.37; 3.6A; 3.6.1;
3.9.1, 3.9.2, 3.9.3; 4.4.3;
5.2.5.1.4; 6.3; 7.3.1.1.1;
7.3.1.1.4; 9.3.6; 9.5; 15

FSAR Tables:

3.2-1;

Technical Specifications (including bases)

3/4.3; 3/4.4; 3/4.5; 3/4.6

Technical Requirements Manual

3/4.3, Table 3.6-2

Commitment Management System

LBDs/Documents reviewed manually:

Keywords:

LPSI, Low Pressure Safety Injection, Void(s), Bubble(s), LOCA, full of water, RWSP, HYTRAN, SYSFLO, PIPER

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
x No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	■	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	■	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	■	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	■	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	■	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	■	Discharge any chemicals new or different from that previously discharged?
7.	☐	■	Change the design or operation of the intake or discharge structures?
8.	☐	■	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	■	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	■	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	■	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	■	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	■	Involve the installation or modification of a stationary or mobile tank?
14.	☐	■	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	■	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	■	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	■	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	■	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	■	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	■	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	■	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	■	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	■	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	■	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	■	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) _____ ☐ No

Name of Security Plan reviewer (print) / Signature / Date

V. 50.59 EVALUATION EXEMPTION **NOT REQUIRED**

Enter this section only if a "yes" box was checked in Section II.A, above.

A. Check the applicable boxes below. If any of the boxes are checked, a 50.59 Evaluation is not required. If none of the boxes are checked, perform a 50.59 Evaluation in accordance with Section VI. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.6.1.1:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.6.1.2. Reference 50.59 Evaluation # _____ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.6.1.3. Reference: _____

☐ The proposed activity is controlled by another regulation per Section 5.6.1.4.

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10CFR50.59 Review Program Guidelines for guidance.

NOT REQUIRED

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:
ER-W3-2003-0112-000 evaluates the LPSI train "B" piping with the presence of voids (up to 0.8 ft^3) in the entire LPSI B train. This condition is being incorporated into the design of the LPSI Train "B" discharge piping. The new void limit will revise the existing limit in OP-903-026. Technical Specification Basis 3/4.5.2 and 3/4.5.3 are revised to address the void, which is applicable to LPSI train "B" discharge piping only (DRN 03-0445). ER-W3-2002-0468-000 performed a similar evaluation for the "A" train

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Reason for proposed Change:

Technical Specification 3/4.5.2 surveillance requires verification at least every 31 days that the ECCS is full of water. Due to potential leakage past containment isolation valves (SI-142B/143B), nitrogen gas has been coming out of solution and forming gas pockets at certain locations in the SI system (Ref. CR-W3-2002-0818). A design basis void will ensure the system remains operable and within design allowables when small voids are in the system.

Currently, the presence of any size gas void is considered to be an adverse condition. ER-W3-2003-0112-000 will allow for "design basis" voiding of up to 0.8 ft^3 in the discharge piping. This void is the total allowable for the entire LPSI train B. The maximum void size to support system operability is not changing as a result of this ER.

50.59 Evaluation summary and conclusions
The evaluation determined that the frequency and consequences of an accident evaluated in the FSAR are not impacted. In addition the likelihood and consequences of a malfunction of the LPSI system is not impacted. The piping system was evaluated for the effects of a hydraulic transient loading and the piping meets the applicable Code requirements. The evaluation meets all applicable design, material and construction standards applicable to the LPSI train B system. No physical work is required. There is no change to the Operating License. The Technical Specification Bases are being revised to address that a void may be present (DRN 03-0445). Prior NRC approval is not required for this change.

B. License Amendment Determination

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 FSAR?

☐ Yes
■ No

BASIS:

The Accident analyses in the FSAR were reviewed and it was determined this proposed change will not impact the frequency of an occurrence of any accident. The LPSI system is not an initiator for any accidents described in the FSAR. During normal operation the LPSI system is aligned to support automatic initiation following a LOCA or MSLB. During normal operation, the LPSI pumps are off and the LPSI header to RCS loop control valves are closed (i.e., system is in standby mode). The system is used to mitigate the effects of an accident involving a decrease in reactor water inventory. There is no increase in the possibility of an RCS leak. There is no change to the pressure boundary of the LPSI system. The system has been evaluated for the hydraulic transient loading and the temporary pressure spike associated with the presence of voids during pump start.

A hydraulic transient could occur during pump surveillance if the pump is started with a void in the line. By limiting the void size to a total of 0.8 ft^3, the pressure in the line will never exceed the design pressure and the pipe stress is still below Code allowables.

Therefore, this ER will not result in more than a minimal increase in 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 FSAR?

☐ Yes
■ No

BASIS:

The LPSI system is part of the Emergency Core Cooling System (ECCS). It is designed to provide core cooling in the event of a LOCA or Main Steam Line Break (MSLB). The LPSI system also provides Shutdown Cooling functions. There is no increase of a likelihood of a malfunction of the LPSI system. The LPSI Train B piping has been evaluated for the presence of some voids and found acceptable. All supports and structural steel have been evaluated and found acceptable.

The pressure in the system remains within the existing design value. There will be no inadvertent lifting of any pressure relieving devices with a total void of 0.8 ft^3 in the LPSI train B. All instrumentation has been found acceptable for the effects of the hydraulic transient. With the presence of voids, the system will still be able to deliver its required flow.

Technical Specification Bases B 3/4.5 provides the bases for ensuring the systems are full of water. Being full of water ensures that there will not be potentially damaging hydraulic transient, pump cavitation or non-compressible gas from entering the reactor. With voids not exceeding a total size of 0.8 ft^3 in the injection legs of LPSI B, the piping is qualified for the hydraulic transient loading. Stresses remain below the approved code allowables. Voids in the discharge section of the pipe will not cause pump cavitation. The void is traveling away from the pump. Lastly, the small nitrogen void has been evaluated if it enters the reactor and it was found negligible. DRN 03-0445 was issued to revise Tech. Spec. bases to add a statement about the void for the discharge piping in LPSI train B.

Information Notice 2002-018 documented an Auxiliary Feedwater Pump failure event at Callaway that was attributed to gas coming out of solution near the pump impeller. The Callaway pump takes suction from a closed Condensate Storage Tank that is sparged with nitrogen. Waterford 3 personnel reviewed the applicability of this event to Waterford 3 equipment via operating experience condition report LO-OPX-2002-00146. The evaluation concluded, in part, that the Waterford 3 Safety Injection system is not susceptible to this phenomenon. The SI pumps take suction from the Refueling Water Storage Pool, which is vented to the atmosphere. Because it is vented, there will not be any significant amount of dissolved gas in the fluid being pumped during the injection mode of operation. The voids that are forming in the LPSI discharge piping are a result of leakage of nitrogen saturated water from the Safety Injection Tanks past check valves SI-142B and SI-143B. The nitrogen saturated water cannot migrate back to the pump through the discharge piping, because the discharge piping immediately downstream of the pump is opened to the RWSP and is thus depressurized. Nitrogen coming out of solution naturally collects at the system high point immediately upstream of the flow control valves near vent valves SI-133B and SI-134B. After receipt of a Recirculation Actuation Signal, the HPSI and CS pumps' suction switches from the RWSP to the Safety Injection Sump and LPSI is secured. This sump is open to the containment atmosphere, which is limited to 44 psig. Therefore, no significant amount of nitrogen will remain in solution during this mode of operation.

During shutdown cooling, a void may enter the pump suction during recirculation. By limiting the void to 0.8 ft^3, the void fraction of the gas and water mixture will be limited so the pump will not be adversely affected. In addition, the Reactor Coolant System provides suction to the LPSI pumps during this mode. The significantly higher NPSH provided by the RCS will counter-act the effects of gas coming out of solution at the pump during Shutdown Cooling. Also, plant operating experience has demonstrated the satisfactory operation of the LPSI pumps during Shutdown Cooling. Because gas saturated solution will not adversely effect the pumps' performance during injection and recirculation modes, and it will have no adverse effect during shutdown cooling mode, the current NPSH calculations for the SI pumps remains valid, and there is no need to adjust for gas saturation. Therefore, operation with gas voids of 0.8 ft^3 or less and/or nitrogen saturated fluid in the LPSI discharge piping will not increase the likelihood of occurrence of a malfunction of any structure system or component important to safety as evaluated in the FSAR.

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

BASIS:

The Accident Analyses in the FSAR were reviewed and it was determined that this modification will not impact the consequences (radiation dose) of any accident. The safety injection system is designed to ECCS criteria to ensure core cooling in the event of an LOCA or MSLB. The safety injection system (SIS) fluid shall provide sufficient neutron absorbers (boron) to maintain the reactor sub critical. The presence of a minimal void in the discharge piping in train B of LPSI will not impact the safety function of the system. The boron comes from the safety injection tanks (SIT) and the Refueling Water Storage Pool (RWSP). There is no change to the existing boron concentration levels.

Based on an evaluation by Westinghouse (formerly ABB Combustion Engineering) performed in response to CR-WF3-1996-1965, it is concluded that the additional Nitrogen, injected into the core (up to 5 ft^3) will not have an impact on the safety analysis. The total Nitrogen void for the LPSI system is 2.2 ft^3 .This accounts for the Nitrogen in LPSI"A" train as well as the "B" train. The Westinghouse response concludes that the LOCA accident analyses were not significantly impacted due to the addition of all of the Nitrogen initially present in the Safety Injection Tanks into the core. Because the 0.80 ft^3 void in LPSI B train is being allowed by this evaluation is negligible compared to the amount of nitrogen in each of 4 SITs, it is enveloping and therefore the voiding will have no impact on the safety analysis.

The Containment Water Level calculation, MN(Q)6-4, does not credit the volume of the ECCS piping. Therefore, initial voiding in the LPSI piping will have no effect on the calculated safety injection sump level, and will therefore have no adverse effect on the Net Positive Suction Head available to the ECCS pumps post-RAS.

The LPSI pumps are designed to inject large volumes of water at low pressure. The pumps are sized based on the requirements of shutdown cooling, which is more demanding than LPSI injection. A small void in the line will have negligible impact on the ability of the pumps to provide their intended function. The LPSI Train B is still within the design basis with the presence of void up to 0.80 ft^3 in the system will still mitigate the effects of an accident involving a decrease in reactor water inventory. The pressure in the system will not exceed the design pressure of the piping. The pipe supports and associated structural steel meets the appropriate design allowables due to the hydraulic transient loading.

Therefore, this ER will not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

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

BASIS:

This design void to the LPSI system will not increase the consequences (radiological release) of a LOCA or MSLB. There is no increase in consequences of LOCA or MSLB. The presence of voids in the discharge piping of LPSI train "B" will cause hydraulic transient loading and a pressure spike in the system. The piping has been evaluated for this transient and the piping remains within the ASME Code allowables. The pressure spike will not exceed the current design pressure of the system. The pipe supports have also been evaluated and found to be within their design allowables.

The hydraulic transient loading will not have any adverse effect on the system. The loading on the pump nozzles are still within the vendors design allowables. The loading on the containment penetration is still within the design allowables. The hydraulic transient will not subject any system valves to pressure locking. The additional nitrogen that would be injected into the core following a LOCA has been evaluated and found to have a negligible impact. There is no greater reliance placed on any other system or structure due to this evaluation.

Therefore, this ER will 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 FSAR

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

BASIS:

This design void to LPSI B train does not create an accident of a different type other than the previously evaluated accidents. There are no new system interfaces created. The LPSI system is not an initiator for any accidents. The system is normally in standby mode. Any voids in the system while the system is in standby mode will have no effect on any other system. During shutdown cooling, the void is limited which will prevent pump binding. This will ensure no unnecessary burden on the system during shutdown cooling (SDC). In addition, in SDC the LPSI pumps are started with the recirculation isolation valve open. Pressure will never build in the system so it is not susceptible to a hydraulic transient. The piping, piping components, valves and penetrations have all been evaluated for the effects of the hydraulic event.

Therefore, this ER will no create a possibility for an accident of a different type than any previously evaluated in the FSAR

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

BASIS:

The design void will not impact the operation or performance of the LPSI system. The LPSI will not cause a malfunction of any SSC. The LPSI system was designed to meet its functional requirement with even the failure of a single active component. The piping, pipe supports and building steel are acceptable for the hydraulic transient loading.

The presence of a 0.80 ft^3 (total) void in the LPSI B train will not impact the ability of the system to deliver its rated flow. In SDC the recirculation line is open so any void in the pipe will be directed to the suction of the LPSI pump initially. By limiting the void size will not adversely affect the pump performance. The hydraulic transient will not interfere with any other system ability to perform their safety function. The operation of the system is not affected by this modification. The loading on the pump due to the transient will not effect the pumps ability to deliver is rated capacity.

Therefore, this ER will 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 FSAR

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

BASIS:

This evaluation will allow gas pockets totaling up to 0.80 ft^3 to be present at penetration 36, between containment isolation valves SI-143B (inside containment check valve) and SI-139B and at penetration 37, between SI-142B (inside containment check valve) and SI-138B. Events requiring containment isolation also require safety injection actuation. Therefore, the void will be collapsed or swept out of the system when the SI system becomes functional following SIAS. In addition, Penetrations 36 and 37 are exempt from Type C Leak testing because safety injection flow can be guaranteed to exist under all LOCA conditions, even considering single failure criteria (FSAR Table 6.2-43). If the LPSI pump were to fail to start, then flow from the HPSI header would prevent containment atmosphere from leaking through the break and into the low-pressure header. Therefore, the presence of 0.80 ft^3 voids at either penetrations 36 and 37 will not compromise containment integrity. Also, the void will not prevent closure of any of the containment isolation valves. Check valves SI-142B and SI-143B are normally closed because the Safety Injection Tanks provide higher pressure downstream of the check valves than the static head provided by the RWSP on the upstream side. Furthermore, if there were no differential pressure across the check valves, there would be no mechanism for gas to come out of solution. Voiding will therefore not affect the ability of SI-142B or SI-143B to perform their containment isolation functions. Valves SI-138B and SI-139B are located at relative low points in the system, where gas will not collect. Therefore, voiding will not affect the operation of these valves.

Therefore, this ER will not impact a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

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

BASIS:

The FSAR does not provide detail on which systems are to be evaluated for a hydraulic transient. Currently Waterford has performed hydraulic transient evaluations for system that are susceptible (i.e., ACCW and EFW). The Waterford 3 pipe stress design guide (AMEC-D-001) states *"A time history analysis shall be performed using the available computer program along with the proper forcing functions (force versus time)."* There is no specific guidance given on the program to be used to develop the time history input (definition of the hydraulic transient event). In the case of the LPSI analyses, ME101 was used to perform the time history analyses. This program is described in the Section 3.9.1.2.1.11 of the FSAR.
To generate the hydraulic transient loading, time histories HYTRAN, computer program was used. This program is described in FSAR section 3.9.1.2.1.19

Therefore, this ER will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 Steam Electric Station

Document Reviewed: ER-W3-2002-0675 **Change/Rev.** 0

System Designator(s)/Description: Boric Acid Makeup Tank (BAMT)

Description of Proposed Change

The purpose of ER-W3-2002-0675 is to achieve RCS refueling chemistry conditions sooner. This will be accomplished by changing the appropriate procedure, determined by Operations. By raising one Boric Acid Makeup Tank (BAMT) concentration to a value of 10,000 ppm (~5.7 wt%), Waterford 3 will be able to achieve Reactor Coolant System (RCS) refueling chemistry requirements (Acid reducing and refueling concentration) sooner. Technical Specification (T.S.) sections 3.1.2.7 and 3.1.2.8 require one BAMT boric acid concentration to be maintained between 2.25 wt% and 3.5 wt% (3.5 wt% = 6125 ppm) during modes 1 through 6. It is desired to increase the boron concentration in one BAMT, declared inoperable, to 10,000 ppm. This will reduce the time required for the RCS to reach the required acidic conditions to support chemistry and to achieve refueling concentration. T.S. section 3.1.2.2 requires two boron injection flow paths to the RCS via the charging pumps, one flow path from an acceptable BAMT via the BAM pump and the other flow path from an acceptable BAMT via its gravity feed valve. This T.S. is maintained via the operable BAMT utilizing its associated BAM pump and gravity feed valve. It is expected that this will be a temporary condition just prior to shutdown for a refueling outage.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert Item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: 03-010)	Sections I, II, III, IV, and VI required

Preparer: Richard K. Baird / Richard K. Baird / EOI / NE / 7/23/03
Name (print) / Signature / Company / Department / Date

Reviewer: Edmond Wiegert / Edmond D. Wiegert / EOI / NE / 7-23-03
Name (print) / Signature / Company / Department / Date

OSRC: K. WALSH [signature] 8/12/03
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☐	☒	
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The purpose of ER-W3-2002-0675 is to allow raising one BAMT boron concentration to a value of 10,000ppm (5.7%). Technical Specification (T.S.) sections 3.1.2.7 and 3.1.2.8 require one BAMT boron concentration to be maintained between 2.25% and 3.5% during modes 1 through 6. T.S. section 3.1.2.2 requires two boron injection flow paths to the RCS via the charging pumps, one flow path from an acceptable BAMT via the BAM pump and the other flow path from an acceptable BAMT via its gravity feed valve.

FSAR Table 6.3-5A, Core and System Parameters Used in the Long Term Cooling Analysis, lists the initial boric acid concentration for the BAM tanks at 3.5%. The purpose of this analysis is to determine the acceptability of raising the applicable or necessary limits. It performs this function by analyzing when boric acid precipitation will occur following a LOCA where the RWSP, SI tanks, and BAM tanks are injected into the RCS.

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LRS Fullfind search:	See above for basis
LBDS_50.59 search performed:	"BAMT" 8 hits / 1 relevant "BAMU" 3 hits / 1 relevant "Boric Acid Makeup" 27 hits / 2 relevant "LTC" / "Long Term Cooling" 8 hits / 1 relevant "shutdown procedure" 5 hits / 0 relevant "boron concentration" 5 hits / 0 relevant "RCS concentration" 1 hit / 0 relevant
Technical Specification Sections 3.1.2.7 and 3.1.2.8, FSAR Figure 3.1-1, FSAR Sections 1.7, 1.9, 3.11, 3.2, 3.9, 6.3, 7.4, 7.7, 9.3, 11.3, 11.5, 12.2, 14.2, FSAR table 6.3-5A	

LBDs/Documents reviewed manually:

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
☒ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) _____ ☐ No

_____

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

The purpose of ER-W3-2002-0675 is to achieve RCS refueling chemistry conditions sooner. This will be accomplished by changing the appropriate procedure, determined by Operations. By raising one Boric Acid Makeup Tank (BAMT) concentration to a value of 10,000 ppm (~5.7 wt%), Waterford 3 will be able to achieve Reactor Coolant System (RCS) refueling chemistry requirements (Acid reducing and refueling concentration) sooner. Technical Specification (T.S.) sections 3.1.2.7 and 3.1.2.8 require one BAMT boric acid concentration to be maintained between 2.25 wt% and 3.5 wt% (3.5 wt% = 6125 ppm) during modes 1 through 6. It is desired to increase the boron concentration in one BAMT, declared inoperable, to 10,000 ppm. This will reduce the time required for the RCS to reach the required acidic conditions to support chemistry and to achieve refueling concentration. T.S. section 3.1.2.2 requires two boron injection flow paths to the RCS via the charging pumps, one flow path from an acceptable BAMT via the BAM pump and the other flow path form an acceptable BAMT via its gravity feed valve. This T.S. is maintained via the operable BAMT utilizing its associated BAM pump and gravity feed valve. It is expected that this will be a temporary condition just prior to shutdown for a refueling outage.

One concern with increased boron concentrations is the potential for flow blockage resulting from significant precipitation. To address this concern, procedural restrictions have been established and a supporting field test has been performed. The procedural restrictions are: a) this portion of the procedure is only allowed at EOC (>341 EFPD) with RCS boron concentration ≤ 300ppm, b) the designated BAMT that will be declared inoperable should not be batched up until ≤7 days prior to expected usage, c) if a stagnant condition occurs in the piping, or is anticipated to occur, for >1 hour with highly concentrated boric acid present, the piping must be flushed with boric acid concentration of ≤ 3.5wt%, or temperature must be maintained at sufficient temperature (>80°F), or flow must be established (recirculation) so boric acid remains in solution, d) following the injection of the highly concentrated boric acid to the RCS, the piping must be flushed with a boric acid concentration of ≤ 3.5wt%, e) if the low temperature alarm is received for the elevated boric acid concentration BAMT (90°F), Operations must take action to restore temperature in the BAMT to ≥90°F, if temperature cannot be maintained ≥90°F, flow in the piping between the BAMT and BAM pump must be flushed with a boric acid concentration of ≤3.5wt % or the system must be placed on recirculation, and f) in order for the BAMT with the elevated boron concentration to be returned to service, the BAM tank piping and BAMT level instrument must be flushed with a boric acid concentration of ≤ 3.5wt% and a boron sample from chemistry must be performed to verify that the BAM tanks and piping are within the limits of T.S. sections 3.1.2.7 and 3.1.2.8, as applicable. A field test was conducted on 3/19/2003 to address the crystallization/precipitation issue within the BAM system piping. The results indicate that after 90 minutes a 10,000 ppm solution in a graduated cylinder had no crystalline or precipitation formation. A 15,000 ppm solution had a small amount of crystalline formation but could be easily poured. The test was initiated with the solution initially at 105°F and cooling to 85°F at the 90 minute mark. The solutions were placed in the -4 RAB vice the -35 RAB since the -4 RAB had a colder ambient temperature. Therefore, even if a small amount of boron were to precipitate in the piping, it would not block flow or damage equipment. Also, the U.S. Borax Technical Data for Boric acid solubility in water lists 80°F as the temperature at which boron crystallization will occur for a 10,000 ppm boric acid solution. As part of the test, the glass cylinders were left in the lab after the initial time period test was completed. One week after the initial test was completed, it was observed both solutions could still easily be swirled and poured. The amount of crystalline/precipitation formation after a week was about 5 grams for the 10,000 ppm solution and 15 grams for the 15,000 ppm solution. Comparing the conditions which existed for the field test and the conditions which will be present in the BAM system ensures the field test was conservative: a) The length of the test was 7 days, but Operations is required to

initiate corrective actions (as stated above) within 1 hour to ensure significant precipitation will not occur. b) The field test did not simulate flow or an external heat source. The BAMT contains heaters which are energized to maintain BAMT temperature between 100 and 110°F. Also, since there will be no time > 1 hour with no flow and the increased boron concentration present, the BAMT placed on recirculation will allow its associated pump to add heat. c) The field test was performed in a glass beaker, which is not as efficient of a conductor of heat as stainless steel. The stainless steel piping and tank within the BAM system is lagged and contains an external heat source (BAMT heaters); therefore, this effect is determined to be insignificant. The BAM system will be under static pressures which the field test did not simulate. However, pressure has no effect on the solubility of boric acid (verified with chemistry department). In conclusion, the field test results are conservative to actual plant equipment and flow blockage of the piping will not occur.

Another concern is the clogging of instrument lines within the Boric Acid Makeup or Charging systems. The instrument lines in the BAM system are impulse lines, meaning that the pressure of the system is felt on the instrument. The increased boron concentration would migrate down the instrument lines over a period of time, but only the density of the fluid would have an effect. The two instruments in question are the BAMT level instrument and the charging system flow meter. The BAMT level instrument effect is negligible with the change in the span of the instrument being only 0.08%. The indications produced from this instrument are sufficient with the current calibration. But, following injection of the high boric acid concentration, Operations will be required to back fill or flush only the BAMT level prior to returning the system to an operable condition. The charging flow instrument error is -1% of the reading from the meter. For example, for a 50 GPM flow rate, an additional error of -0.5 GPM would exist (0.5 GPM less than indicator reading). But, since charging flow will continue following the injection of the high boric acid concentration water (the supply will then become the VCT which has a boric acid concentration <3.5wt %), the charging flow meter is not required to be flushed.

The Technical Specification sections 3.1.2.7 and 3.1.2.8 require one BAMT to have a boric acid concentration between 2.25 wt% and 3.5 wt%. Thus, the second BAMT's boric acid concentration may be raised above 6125 ppm (3.5wt %) provided Operations declares the BAMT inoperable before boric acid concentration exceeds 6125 ppm and isolates the tank to minimize the probability of it being injected due to an automatic actuation. The injection of the boric acid into the RCS from the inoperable BAMT has been evaluated and determined to be acceptable. The alignment of the inoperable BAMT to the RCS does not adversely affect the integrity of the RCS and does not adversely affect the safety function of the operable BAMT. Therefore, the safe operation and the design basis of the plant are maintained.

T.S. section 3.1.2.2 requires two boron injection flow paths to the RCS via the charging pumps, one flow path from an acceptable BAMT via the BAM pump and the other flow path form an acceptable BAMT via its gravity feed valve. This T.S. is maintained via the operable BAMT utilizing its associated BAM pump and gravity feed valve.

FSAR Table 6.3-5A, Core and System Parameters Used in the Long Term Cooling Analysis, lists the initial boric acid concentration for the BAM tanks at 3.5%. The purpose of this analysis is to determine when boric acid precipitation will occur following a LOCA where the RWSP, SI tanks, and BAM tanks are injected into the RCS. This analysis assumes an initial RCS boron concentration of 3000 ppm and an initial BAMT boron concentration of 6125 for each tank. Since this change will only be in effect at End of Cycle (EOC) for refuel purposes, the actual RCS boron concentration will be 0 ppm or a value less than that assumed in the calculation. Therefore, utilizing a conservative estimate for actual RCS boron concentration (300ppm) and an initial RCS boron concentration of 2000 ppm as assumed in the analysis for this evaluation, the total RCS boron concentration, following injection of the RWSP, SI tanks, and one BAMT at 6125 ppm and the other at 10,500 ppm, is less than that used in the analysis. Therefore, the analysis is bounding and no change to the FSAR will occur.

Reason for proposed Change:
The purpose of ER-W3-2002-0675 is to achieve RCS refueling chemistry conditions sooner. This will be accomplished by changing the appropriate procedure(s) as determined by the Operations Department. By raising one Boric Acid Makeup Tank (BAMT) concentration to a value of 10,000 ppm (~5.7 wt%), Waterford 3 will be able to achieve Reactor Coolant System (RCS) refueling chemistry requirements (Acid reducing and refueling concentration) sooner. Technical Specification (T.S.) sections 3.1.2.7 and 3.1.2.8 require one BAMT boric acid concentration to be maintained between 2.25 wt% and 3.5 wt% (3.5 wt% = 6125 ppm) during modes 1 through 6. It is desired to increase the boron concentration in one BAMT, declared inoperable, to 10,000 ppm. This will reduce the time required for the RCS to reach the required acidic conditions to support chemistry and to achieve refueling concentration. T.S. section 3.1.2.2 requires two boron injection flow paths to the RCS via the charging pumps, one flow path from an acceptable BAMT via the BAM pump and the other flow path form an acceptable BAMT via its gravity feed valve. This T.S. is maintained via the operable BAMT utilizing its associated BAM pump and gravity feed valve. It is expected that this will be a temporary condition just prior to shutdown for a refueling outage.

50.59 Evaluation summary and conclusions

This change does not add nor remove any equipment from the plant. Thus, there is no impact to the frequency of accidents or malfunction of equipment. This change does not alter the function of the BAM system or the way that it maintains the ability to maintain shutdown margin. This change does not affect any fission product barrier or method evaluation. Appropriate administrative controls are in place to ensure that the BAMT with the elevated concentration will be declared inoperable while preserving the Technical Specification requirements. Appropriate controls are also in place to ensure that boron crystallization will not occur in either the BAMT or associated piping. The consequences of FSAR Chapter 15 accident analyses are unchanged and bounded by the current analyses. Therefore, it is concluded that prior NRC approval of this change is not required.

B. License Amendment Determination

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 FSAR?

   ☐ Yes
   ☒ No

   BASIS: **ER-W3-2002-0675 allows raising one BAMT concentration to 10,000 ppm to accelerate achieving Reactor Coolant System (RCS) refueling chemistry requirements. The alignment of this BAMT does not adversely affect the integrity of the RCS pressure boundary. There is no impact to any accident initiator related to the charging or boric acid makeup systems as a result of this change. Therefore, there is no increase in frequency of occurrence of an accident.**

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 FSAR?

   ☐ Yes
   ☒ No

BASIS: Because this ER increases the boron concentration within the BAM system, crystallization is a potential concern. A simulation was conducted on 3/19/03 to address the crystallization issue within the BAM system piping. The results indicate that after 90 minutes a 10,000 ppm solution in a graduated cylinder had no crysatlline formation. As part of the test, the glass cylinders were left in the lab after the test was completed. The week after the test was completed, both solutions could still easily be swirled and poured. To further reduce the potential for precipitation, Operations will be required to maintain piping temperature at a sufficient temperature (80°F for 10,000 ppm) to ensure boric acid remains in solution, flush the piping with a water concentration of ≤ 3.5wt%, or establish flow in that piping if a stagnant condition exists, or is anticipated to occur, in the piping for >1 hour with the elevated boric acid concentration present.

Another concern is the clogging of instrument lines within the Boric Acid Makeup or Charging systems. The instrument lines in the BAM system are impulse lines, meaning that the pressure of the system is felt on the instrument. The increased boron concentration would migrate down the instrument lines over a period of time, but only the density of the fluid would have an effect. The two instruments in question are the BAMT level instrument and the charging system flow meter. The BAMT level instrument effect is negligible with the change in the span of the instrument being only 0.08%. The indications produced from this instrument are sufficient with the current calibration. But, following injection of the high boric acid concentration, Operations will be required to back fill or flush only the BAMT level instrument prior to returning the system to an operable condition. The charging flow instrument error is -1% of the reading from the meter. For example, for a 50 GPM flow rate, an additional error of -0.5 GPM would exist (0.5 GPM less than indicator reading). But, since charging flow will continue following the injection of the high boric acid concentration water (the supply will then become the VCT which has a boric acid concentration <3.5wt %), the charging flow meter is not required to be flushed. While the RCS is lined up to the inoperable BAMT, the potential for RCS inventory loss exists. But, because of the check valves within the charging system, inventory loss will not occur. The charging system contains check valves to each loop connection and the pressurizer aux spray line. Also, there are check valves at each charging pump discharge. For the BAM system, check valves are within the system at the BAM pump discharges, as well as between the BAM common discharge header and the charging pump suction header. Therefore, RCS inventory loss will not occur.

Therefore, this change will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

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

☐ Yes
☒ No

BASIS: The Post LOCA Long Term Cooling Analysis purpose is to determine when boric acid precipitation will occur following a LOCA where the RWSP, SI tanks, and BAM tanks are injected into the RCS. This analysis assumes an initial RCS boron concentration of 3000 ppm and an initial BAMT boron concentration of 6125 ppm for each tank. Since this change will only be in effect at End of Cycle (EOC) for refuel purposes, the actual RCS boron concentration will be 0 ppm. Therefore, utilizing a conservative estimate for actual RCS boron concentration (300ppm) and an initial RCS boron concentration of 2000 ppm, the total RCS boron concentration, following injection of the RWSP, SI tanks, and one BAMT at 6125 ppm and the other at 10,500 ppm, is less than that used in the analysis. Therefore, since the analysis is bounding and Technical Specification compliance is maintained, no increase to the consequences of an accident will occur.

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 FSAR? ☐ Yes ☒ No

BASIS: This change will allow raising one BAMT boric acid concentration to 10,000 ppm, and it will be declared inoperable. By declaring the BAMT with the elevated boric acid concentration inoperable, the T.S. are met via the operable BAMT (maintained in accordance with Figure 3.1-1, Required Stored Boric Acid Volume as a Function of Concentration), its gravity feed valve, and associated BAM pump. Also, since the total amount of boron is unchanged (see discussion in question #3 above) for the containment sump pH analysis, the amount of tri-sodium phosphate required to maintain containment sump pH following a LOCA is unchanged. Therefore, there is no increase in the consequences of any malfunction.

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

BASIS: This change does not add or remove any equipment from the plant. This change will accelerate achieving RCS refueling chemistry requirements. The BAM system is still capable of and will perform its design function. Therefore, this change does not create the possibility for an accident of a different type.

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 FSAR? ☐ Yes ☒ No

BASIS: This change does not add or remove any equipment from the plant. This change will accelerate achieving RCS refueling chemistry requirements. There are no new failure modes with different results than any previously evaluated in the FSAR. Therefore, this change does not create the possibility for a malfunction with different results.

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

BASIS: This ER does not affect any fission product barrier. If a LOCA were to occur while injecting this high concentration of boric acid, the effects would be as described in the Post LOCA Long Term Cooling Analysis. This analysis envelopes the boron precipitation since it assumes worst case initial RCS boric acid concentration, which occurs at the beginning of cycle. This change is only allowed at end of cycle for refueling outages and therefore the long term cooling analysis envelopes the boron precipitation. Therefore, this change will not result in a design basis limit being exceeded.

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

BASIS: The Technical Specification sections 3.1.2.7 and 3.1.2.8 require only one BAMT to maintain a boric acid concentration between 2.25 wt% and 3.5 wt%. This change only allows declaring one BAMT inoperable. This BAMT is allowed to have an elevated boron concentration in excess of 3.5wt%. T.S. section 3.1.2.2 requires two boron injection flow paths to the RCS via the charging pumps. This change only allows declaring one BAMT inoperable, therefore the T.S. is met utilizing the operable BAMT, its associated gravity feed valve, and applicable BAM pump. Therefore, no methods of evaluation described in the FSAR are changed.

I. OVERVIEW / SIGNATURES

Facility: WATERFORD 3

Document Reviewed: ER-W3-2002-0283-002 **Change/Rev.** 0

System Designator(s)/Description: SDC , SI

Description of Proposed Change

A portion of the Shutdown Cooling (SDC) suction piping for Train A and B remains empty when the system is in the standby mode. The section of SDC suction piping is located between Containment Isolation Valves (CIV) SI-405A(B) and SI-407A(B) at Penetrations 40 and 41. ER-W3-2002-0283-002 changes the design basis for the SDC system to reflect Penetrations 40 & 41 as air-filled penetrations and therefore a containment bypass leakage pathway subject to 10CFR50 Appendix J Type "C" testing.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert *item # from Section 5.2.2.2*).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
■	50.59 EVALUATION (#: _03-011_____)	Sections I, II, III, IV, and VI required

Preparer: M.L. Touna / Mot Tam / Entergy Operations / Design Engineering 9-30-03
Name (print) / Signature / Company / Department / Date

Reviewer: Chris A. Davenport / [signature] /DP Engineering, Ltd / 9/30/03
Name (print) / Signature / Company / Department / Date

OSRC: K. Peters / [signature] / 10/14/03
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:

II. SCREENING

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	■	
TS	☐	■	
NRC Orders	☐	■	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	■	☐	FSAR Table 6.2-32 & 6.2-43, DRN 03-1014
TS Bases	☐	■	
Technical Requirements Manual	■	☐	TRM Table 3.6-1, DRN 03-1023
Core Operating Limits Report	☐	■	
NRC Safety Evaluation Reports[1]	☐	■	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	■	
Emergency Plan[2]	☐	■	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	■	
Offsite Dose Calculations Manual	☐	■	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ Yes ■ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation? ☐ Yes ☐ No ■ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

II. SCREENING

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

A portion of the Shutdown Cooling (SDC) suction piping for Train A and B remains empty when the system is in the standby mode. The section of SDC suction piping is located between Containment Isolation Valves (CIV) SI-405A(B) and SI-407A(B) at Penetrations 40 and 41.

ER-W3-2002-0283-002 changes the design basis for the SDC system to reflect Penetrations 40 & 41 as air-filled penetrations. The ER determined that the effects on the SDC system and components upon initiation of Shutdown Cooling, due to the initially empty piping, are acceptable. Therefore, based on the ER evaluation, the LBD's that discuss operation and safety analyses of the SDC system or components were reviewed and determined not to be impacted based on the results of the ER. In addition, although the change will require LLRT of Penetrations 40 & 41, the change does not represent a test or experiment not described in the SAR because LLRT is described in the SAR.

The ER also determined that the empty piping between the CIV's at Penetrations 40 and 41 negates their current exemption from 10CFR50 Appendix J Type C leakage testing. The ER requires that the penetrations be added to the 10CFR50 Appendix J Waterford 3 containment leakage-testing program. Therefore, the LBD's that discuss containment integrity were reviewed. FSAR Table 6.2-32 was impacted because it currently states that Penetrations 40 and 41 do not require Type C testing. FSAR Table 6.2-43 is impacted because it currently states the Penetrations 40 and 41 are exempt from Type C leakage testing because the penetration piping will maintain a water seal for 30 days post accident. TRM Table 3.6-1 is impacted because it does not currently include valves SI-405A(B) and SI-407A(B).

Waterford 3 does not utilize an independent spent fuel storage installation.

II. SCREENING

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

Entergy FullFind Software was used to perform an electronic search of the LBD's. The keywords used were intended to identify all LBD's that discuss the operation of the SDC system and components and all LBD's that discuss containment integrity and specifically Penetrations 40 and 41. All LBD's identified by the keyword searches were reviewed manually using the controlled electronic files in IDEAS or controlled hard copies in the library.

LBDs/Documents reviewed via keyword search:	Keywords:
LBDS_50_59	"Shutdown Cooling", "SDC System", "Residual Heat Removal", "RHR System", "penetration piping", "Appendix J", "Type C", "Containment Integrity"

LBDs/Documents reviewed manually:

Safety Evaluation Report (NUREG-0787); Sections 5.2.2.2, 5.2.5, 5.4.3, 6.2 & 7.4.2

Standard Review Plan (NUREG-0800); Sections 5.4.7, Branch Technical Position RSB 5-1

Final Safety Analysis Report; Sections 3.1.30, 3.6.1, 5.2.1.4, Appendix 5.2B, 6.2.4, 6.3.1.4.2, 6.3.2.2.2.1, 6.3.2.9.56.3.3.4, 6.3.5.4.1, 7.4.1.3, 9.3.6, Chapter 15; Tables 6.2-32, 6.2-43, 9.3-16

Technical Specifications; 3/4.3.3.5, 3/4.4.5.2, 3/4.4.8.1, 3/4.4.8.3, 3/4.4.9, 3/4.5.2, 3/4.5.3, 3/4.6.1.2, 3/4.6.3, 3/4.9.8.1, 3/4.9.8.2

Technical Requirements Manual; Table 3.6-1, Table 3.6-2

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
■ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	■	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	■	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	■	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	■	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	■	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	■	Discharge any chemicals new or different from that previously discharged?
7.	☐	■	Change the design or operation of the intake or discharge structures?
8.	☐	■	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	■	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	■	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	■	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	■	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	■	Involve the installation or modification of a stationary or mobile tank?
14.	☐	■	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	■	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	■	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	■	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	■	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	■	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	■	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	■	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	■	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	■	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	■	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	■	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) _____ ☐ No

_____

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

ER-W3-2002-0283-002 changes the design basis for the Shutdown Cooling (SDC) System to reflect Penetrations 40 & 41 as air-filled penetrations and therefore a containment bypass leakage pathway subject to 10CFR50 Appendix J Type "C" testing. The FSAR is being revised to reflect that Penetrations 40 and 41 require Type "C" testing. The TRM is being revised to reflect that Penetrations 40 and 41 are by-pass leakage paths.

Reason for proposed Change

The current design and piping geometry of the SDC system prevents the piping between the CIV's in Penetrations 40 and 41 from remaining full. Penetrations 40 and 41 are currently exempt from 10CFR50 Appendix "J" leakage testing on the premise that the penetrations will maintain a water seal for 30 days post accident. This change will reconcile the design basis of Penetrations 40 and 41 to be consistent with the physical plant.

50.59 Evaluation summary and conclusions

ER-W3-2002-0283-002 required that the CIV's in Penetrations 40 and 41 be added to the 10CFR50 Appendix "J" Waterford 3 containment leakage-testing program. This required a change to the FSAR and TRM which currently reflect the exemption of the CIV's in Penetrations 40 and 41 from the Appendix "J" testing. The 50.59 evaluation demonstrated that a Licensing Amendment is not required because the potential bypass leakage, due to the lack of a water seal, from Penetrations 40 and 41 would be included in the Appendix "J" testing program. Maintaining the penetration leakage within Appendix J program requirements will ensure that total containment bypass leakage, including Penetrations 40 and 41, remains subject to 10CFR50 Appendix J limits. In addition, operation and performance of the SDC and Containment Isolation systems are not affected and therefore safety analysis is not affected. System redundancy and independence is not reduced and no new system interactions are created. No greater reliance on any SSC is created and no new failure modes with different results are created. The design basis limits for fission product barriers are not affected and no new methods for establishing design basis and safety analyses are utilized.

B. License Amendment Determination

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 FSAR?**	☐ ■	Yes No
	BASIS: SDC System Operation The accidents analyzed in the FSAR were reviewed. The accidents analyzed in Chapter 15 are not initiated by the Shutdown Cooling system. However, the following two accidents can be initiated by the Shutdown Cooling (SDC) system. SDC Moderate Energy Line Break: In FSAR Section 9.3.6.3.2 an SDC line break during shutdown cooling operation is discussed. ER-W3-2002-0283-002 demonstrates that adequate design margin exists in the SDC suction lines to accommodate the potential hydraulic transients and still remain within ASME Code limits. Therefore, based on the ER evaluations, there is not more than a minimal increase in the frequency of occurrence of a SDC suction line break. Loss of SDC flow: The loss of SDC flow with the RCS partially filled is discussed in FSAR Section 9.3.6.3.4. ER-W3-2002-0283-002 demonstrates that the presence of empty SDC suction piping does not affect the Failure Mode and Effects Analysis (FSAR Table 9.3-16) for the SDC system. The ER demonstrates that the overall system performance and reliability is not affected, does not cause the system to be operated outside its design limits and does not cause the system to be operated differently. The change does not increase the possibility of an operator error because it does not affect system operation. Therefore, based on the ER evaluations, there is not more than a minimal increase in the frequency of occurrence of a loss of SDC flow. Containment Isolation The Shutdown Cooling suction Containment Isolation valves in Penetrations 40 and 41 are not accident initiators.		

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 FSAR?**	☐ ■	Yes No

BASIS:

The Structure, System or Components (SSC) important to safety that are located in the SDC system and could potentially be affected by the presence and passage of an air pocket are as follows; Low Pressure Safety Injection (LPSI) pumps A & B, SDC flow elements, SDC suction piping and supports, Containment Isolation Valves (CIV) SI-405A(B) and SI-407A(B), Relief valves SI-406A(B) and SI-408A(B).

The LPSI pumps inject water from the RWSP into the RCS during a LOCA. ER-W3-2002-0283-002 demonstrates that the air pocket could potentially grow during a large or small break LOCA. However, the air pocket would not reach the LPSI pump suction during the injection mode. The LPSI pumps are used to initiate SDC flow during normal plant shutdown operations and post-accident to provide long term core cooling. The ER demonstrates that the LPSI pumps would not become gas bound by the air pocket. The ER also demonstrates that the SDC flow could potentially be reduced for only a short period of time (approximately 1 minute) upon initiation of SDC operations as the air pocket passes through the LPSI pumps. The ER demonstrates that the SDC flow instrumentation, SI IFE 1306 & 1307 used for accident monitoring as listed in FSAR Table 7.5-3, would not be affected by the passage of the air pocket through the SDC piping. However, the flow transmitters, SI IFT306 & 307, used to control SDC flow when the controls are put in automatic could potentially be affected and cause oscillations in the flow rate. These transmitters are not safety related and are not required to mitigate an accident. In addition, oscillations in SDC flow would be indicated in the Control Room and an operator would take manual control. By procedure the SDC flow is initiated in manual and is only placed in automatic when the flow has stabilized (Ref. OP-009-005). The ER also demonstrates that the loads on the LPSI pump internals and casings during the passage of the air pocket would not cause any pump damage.

The ER demonstrates that the potential hydraulic transient caused by the compression of the air pocket would not cause piping and support stresses to exceed the ASME code limits. In addition, the potential pressure spike would not cause the relief valves SI-406A(B) or SI-408A(B) to actuate.
The ER demonstrates that the operation of the CIV's SI-405A(B) and SI-407A(B) would not be affected by the presence of an air pocket between the valves. The ER demonstrates that the relief valves SI-406A(B) and SI-408A(B) would operate properly if called upon with an air pocket trapped below the valve seats.

The ER demonstrates that system and component redundancy or independence is not changed. Therefore, based on the ER evaluations, there is not more than a minimal increase in the likelihood of occurrence of a malfunction of the above discussed SSC's.

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

BASIS:

SDC System

The SDC system is required during normal plant shutdown operations and post-accident to provide long term core cooling. ER-W3-2002-0283-002 demonstrates that the SDC flow could potentially be reduced for only a short period of time (approximately 1 minute) as the air pocket passes through the LPSI pumps. This momentary reduction in SDC flow when the system is first put into service would have a negligible impact on core cooling. In addition, the maximum amount of expected air (approximately 14 cubic feet) that would be injected into the Reactor Coolant System (RCS) is negligible compared to the amount of Nitrogen gas (approximately 12,500 cubic feet) that would be injected into the RCS from the Safety Injection Tanks. Also, the loss of shutdown cooling evaluation discussed in FSAR Section 9.3.6.3.4 is not affected. The ER also demonstrates that the SDC flow instrumentation (SI IFE 1306 & 1307) used for accident monitoring as listed in FSAR Table 7.5-3 would not be affected by the passage of the air pocket through the SDC piping. The change does not degrade or prevent actions assumed in the FSAR. Therefore, based on the ER evaluations, there would not be more than a minimal increase in the consequences of an accident.

Containment Isolation System

Penetrations 40 and 41 are currently exempt from 10CFR50 Appendix "J" Containment leakage testing because a water seal was assumed to exist for 30 days post accident. The water seal prevented the potential release of containment atmosphere through the CIV's located in the penetrations. However, it was determined that a water seal cannot be maintained and any leakage of containment atmosphere through the CIV's would be considered by-pass leakage. ER-W3-2002-002 required that the penetrations be added to Waterford 3 Appendix "J" Containment leakage testing program. Maintaining the penetration leakage within Appendix J program requirements will ensure that total containment bypass leakage, including Penetrations 40 and 41, remains subject to 10CFR50 Appendix J limits. The change does not affect the function of the CIV's. Therefore, there would not be more than a minimal increase in the consequences of an accident.

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 FSAR?**	☐ ■	Yes No

BASIS:

Safety Injection System

ER-W3-2002-0283-002 demonstrates that the presence of empty SDC suction piping does not affect the operation of the LPSI pumps during injection mode or any components or instruments required for accident mitigation during a large or small break LOCA. The ER also demonstrates that the Failure Mode and Effects Analysis (FSAR Table 6.3-1) for the Safety Injection System is not affected. Therefore, no greater reliance is placed on any SSC. Based on the ER evaluations, there is not more than a minimal increase in the consequences of a malfunction of the Safety Injection system and components important to safety.

SDC System

There are two independent safety related trains of SDC only one of which is required to mitigate the consequences of an accident. ER-W3-2002-0283-002 demonstrates that the presence of empty SDC suction piping does not affect the Failure Mode and Effects Analysis (FSAR Table 9.3-16) for the SDC system. Therefore, based on the ER evaluations, there is not more than a minimal increase in the consequences of a malfunction of the SDC system and components important to safety.

Containment Isolation System

The CIV's, SI-405A(B) and SI-407A(B), associated with Penetrations 40 and 41 are normally locked closed valves that do not receive an containment isolation signal. Zero leakage of containment atmosphere was assumed for these penetrations post accident because of the assumed existence of a water seal. The lack of a water seal could potentially allow the leakage of containment atmosphere. ER-W3-2002-0283-002 adds the valves to the Waterford 3 Appendix "J" Containment leakage testing program. Maintaining the penetration leakage within Appendix J program requirements will ensure that total containment bypass leakage remains within 10CFR50 Appendix J limits. The ER demonstrates that the operation of the CIV's is not affected. Therefore, there is not more than a minimal increase in the consequences of a malfunction of the containment isolation valves.

5.	Create a possibility for an accident of a different type than any previously evaluated in the FSAR?	☐ Yes ■ No
	BASIS: SDC System ER-W3-2002-0283-002 modifies the design basis for the SDC system. The ER demonstrates that the design basis change does not introduce any new SDC system interactions, does not create any new failure mechanisms, does not create any new accident initiators, or change the effect of the SDC system on any interfacing systems. Therefore, based on the ER evaluations, no accident of a different type can be initiated by the SDC system. Containment Isolation System Containment isolation system helps mitigate an accident and does not initiate an accident. The ER demonstrates that the design basis change does not create any new failure mechanisms or create any new accident initiators. Therefore, the absence of a water seal in Penetrations 40 and 41 does not create an accident of a different type.	

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 FSAR?	☐ Yes ■ No
	BASIS: SDC System ER-2002-0283-002 demonstrates that the presence of empty piping in the SDC system does not create any new interactions or change the current interactions for any of the SDC components important to safety. The ER demonstrates that the change will not adversely affect any SSC's or change system operation or alter their safety function. Therefore, based on the ER evaluations, no new results of a malfunction of SDC components important to safety are created and the change will not alter the result on an existing change. Containment Isolation System Maintaining the penetration leakage within Appendix J program requirements will ensure that total containment bypass leakage, including Penetrations 40 and 41, remains subject to 10CFR50 Appendix J limits. The ER demonstrates that the change will not adversely affect any SSC's or change system operation or alter their safety function. Therefore, the lack of water seal in the penetrations and potential valve seat leakage will not cause a different result than previously evaluated and the change will not alter the result on an existing change.	

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

BASIS:

SDC System

The SDC system maintains fission product design basis limits by providing long term Reactor Core cooling and Reactor Coolant pressure and temperature control for normal plant shutdowns and post accident. ER-W3-2002-0283-002 determined that the effects on the SDC system and components upon initiation of Shutdown Cooling, due to the initially empty piping, are acceptable. The ER determined that the SDC flow is not adversely affected and the instrumentation required to monitor SDC flow post accident is not affected. So, the long term cooling of the core is not adversely affected. Therefore, based on the ER evaluations, the fission product barrier design basis limits will not be exceeded.

LPSI System

The LPSI system maintains fission product design basis limits by injecting borated water into the RCS to ensure the Reactor Core is not uncovered during a LOCA. The ER demonstrated that the LPSI pump injection flow is not affected by the air pocket in the SDC suction piping such that any design basis limit will be impacted. Therefore, based on the ER evaluations, the fission product barrier design basis limits will not be exceeded.

Containment Isolation System

The Containment Building is a fission product barrier. The lack of a water seal will potentially allow Containment atmosphere leakage through Penetrations 40 and 41. However, the penetrations are being added to the Waterford 3 Appendix "J" containment leakage testing program. Maintaining the penetration leakage within Appendix J program requirements will ensure that total containment bypass leakage, including Penetrations 40 and 41, remains subject to 10CFR50 Appendix J limits. Therefore, the fission product barrier design basis limits for Containment will not be exceeded.

8.	**Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?**	☐ ■	Yes No
	BASIS: ER-W3-2002-0283-002 evaluates the effect of initially empty piping in the suction of the SDC system and its effect on SDC System operation, LPSI system operation and Containment integrity. The ER demonstrated that the effects on the SDC and LPSI system did not affect any Safety Analyses evaluations or any evaluation methodology for demonstrating conformance with design basis. The ER demonstrates that the SDC and LPSI system will continue to perform their safety functions as assumed in the safety analysis and by-pass leakage will remain within Appendix "J" limits. The methodology for evaluating design basis limits of fission product barriers and accident consequences are therefore, not affected. The analytical method of air bubble impact on equipment is not described in the FSAR. The FSAR does not provide details on which systems are to be evaluated for a hydraulic transient. In addition, the FSAR does not provide specific guidance on the program to be used to evaluate hydraulic transients. Therefore, no methods of evaluation described in the FSAR are affected.		

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 Steam Electric Station

Document Reviewed: Waterford 3 Steam Generator Chemical Cleaning (SGCC) Special Test Instruction (STI-W3-2003-005) **Change/Rev.:** 0

System Designator(s)/Description: Blowdown (BD) System, Controlled Ventilation Area System (CVAS), Condensate Makeup (CMU) System, Sump Pump (SP) System, Main Steam (MS) System, Station Air (SA) System, Shield Building Ventilation (SBV) System, Liquid Waste Management (LWM) System, Reactor Auxiliary Building Normal Ventilation System

Description of Proposed Change

Framatome Advanced Nuclear Power, Inc. (FANP) will be performing the proposed secondary-side chemical cleaning of the Waterford 3 steam generators (SG) at the end of Cycle 12 (the proposed cleaning will occur in Mode 5 – Cold Shutdown). The proposed Special Test Instruction (STI) contains formal instructions for Entergy Operations Inc. (EOI) personnel to control plant operations while FANP personnel perform the chemical cleaning process.

This 10CFR50.59 Evaluation is for the implementation portion of the Steam Generator Chemical Cleaning (SGCC) only. It addresses mainly the effect of the cleaning process on Waterford 3 Technical Specifications and Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses. Note that the effect on the safety related Heating, Ventilating and Air Conditioning (HVAC) systems that contain charcoal is addressed in this evaluation. The connection of the FANP chemical cleaning equipment to plant systems was evaluated separately as part of Procedure PMC-004-008 "Installation Procedure Steam Generator Chemical Cleaning Equipment Installation". ER-W3-2003-0366-02 addresses the impact of the Steam Generator Chemical Cleaning Process on Waterford 3 Permanent Plant Instrumentation. ER-W3-2003-0522-00 addresses the impact of SGCC chemicals on Fuel and Reactor Coolant System (RCS) Components due to leakage (through steam generator tube(s)) from the secondary to primary side during the chemical cleaning. The effects on the steam generator internals and the environmental effects are addressed in separate 10 CFR 50.59 Reports; a "Process" Exemption and an "Environmental" Exemption. These separate reports address the following issues:

- Chemical Process (Affect on steam generator Internals and vapor space corrosion – see "Process" Exemption).
- Control Room habitability (see "Environmental" Exemption).
- Tank Farm Operation and non-FSAR Chapter 15 events (tornado and hurricane - see "Environmental" Exemption).
- Decomposition and process by products (Impact on plant personnel - see "Process" and "Environmental" Exemptions).
- Vendor equipment failure modes and effects analysis (see "Environmental" Exemption).
- Test Results of Waterford 3 specific laboratory testing (see "Process" Exemption).
- Process Corrosion monitoring (see "Process" Exemption).
- Eddy Current tube testing following SGCC (see "Process" Exemption).
- Waste Processing (see "Environmental" Exemption).
- Environmental releases (see "Environmental" Exemption).

The Steam Generator Chemical Cleaning process is conducted in three major parts; iron removal (iron step) followed by SG drying followed by copper/lead removal (copper step). Several iron removal steps (removes magnetite deposits) and several copper/lead steps (removes copper and lead deposits) will be performed for each steam generator during the cleaning process. The actual number of iron and copper steps will be determined during the cleaning process based upon sample results. The iron and copper/lead steps are performed on each steam generator in parallel. The iron step, aimed at bulk deposit dissolution, is applied in a modified fill/soak/drain mode with nitrogen sparging for additional solution mixing. Following the iron step, the steam generators will be dried with an air sparge for approximately 24 hours in preparation for the copper step. During the drying step air is blown through the steam generators and out the Atmospheric Dump Valves (ADVs). The humidity of the air exiting the ADVs is monitored with humidity probes placed in the exit plume area of the ADVs. The copper step will consist of a full bundle application with air sparging for mixing. Hydrogen peroxide will be added during the copper solvent injection to provide additional oxygen for the oxidation and dissolution of copper deposits. After the copper step is complete, two low volume rinses (LVRs) (~3,000 gallons per SG) and a full volume rinse (FVR) (~31,000 gallons per SG) will be performed.

The iron removal steps utilize an iron solvent composed primarily of Diammonium Ethylenediaminetetraacetic Acid (EDTA) to remove the magnetite deposits. The iron solvent is introduced into the steam generator through connections made to the Blowdown System (via blind flanges downstream of valves BD-107A and BD-107B and vendor supplied chemical tanks, pumps and hoses). The Reactor Coolant System (RCS) temperature during the iron steps is maintained at approximately 195° F (range of 190° F to 195° F). The nominal pressure on the secondary side of the steam generators will be maintained from 0-60 psig during the chemical cleaning. The maximum secondary side pressure will not exceed 70 psig (per STI-W3-2003-005). The initial solvent level injection will be to 228" above the tubesheet (~12,100 gallons). The solvent will remain at this level for 4-8 hours depending upon the dissolution rate of the iron deposits. After the initial fill to 228" is complete, nitrogen sparging will be implemented via the blowdown line connection at a rate of approximately 300 scfm (acceptable range of 250-400 scfm) for five minutes and then secured for ten minutes. The nitrogen sparging sequence will be repeated throughout the iron step application. A rate of 250-400 scfm, targeting 300 scfm, will provide a turnover of the SG volume in ~2 minutes and provide adequate mixing of the EDTA solution. The atmospheric dump valves (ADV - MS-116A or MS-116B) will remain partially opened during the iron step except as required to pressurize the steam generators for drain down.

Samples will be collected from the steam generators by draining ~2,000 gallons of solvent from the steam generators to the FANP process tanks and collecting a representative sample at the end of the drain sequence and then returning the solvent volume to the SGs after the sample is completed. Iron Step samples will be taken at various intervals starting at 2.5 hours after the start of the iron step and continuing throughout the entire 40 hour iron step. The final 40 hour drain down sample will be a composite sample from both SGs collected at 3 different levels during SG drain.

When there is indication that iron dissolution has begun to slow at the initial fill level or insufficient free EDTA remains for further dissolution effectiveness, the solvent will be drained from the SGs and reinjected to the final iron solvent fill level of 387" above the

tubesheet (~31,000 gallons of solvent). Draining and refilling the steam generators will require approximately 4-5 hours. An additional drain and refill cycle will be performed after ~14 hours of total exposure. Nitrogen sparging will be implemented via the blowdown line connection at a rate of approximately 300 scfm for five minutes and then secured for ten minutes as previously stated. The sparging sequence will be repeated throughout the iron step application. As discussed above, samples will be collected from the individual SGs throughout the iron step application alternating between SGs. After ~36.5 hours of total exposure final drain down of the SGs well be initiated. Drain down will require ~3.5 hours to complete, resulting in a total qualified exposure time of 40 hours at the SG tubesheet. Following completion of the draindown, the plant will cool the RCS to a temperature between 145°F and 155°F, targeting ~150°F.

Following draindown of the iron solvent and cooldown of the RCS to 150°F, an approximate 24-hour drying step will be implemented by sparging air at a rate of 650-1000 scfm per SG. Following the drying step, the RCS will be cooled down further to a temperature of about 110°F, if achievable. A minimum cooldown temperature of ~140°F is required for the copper step application.

Once the steam generator temperature is stabilized, injection of the ~6,000 gallons of copper solvent (composed primarily of Ethylenediamine or EDA), hydrogen peroxide and Ammonium Carbonate (5,800 gallons of copper solvent, 200 gallons of hydrogen peroxide and 18 grams/liter of ammonium carbonate) will be initiated and followed by an additional 5,600 gallons of copper solvent. After the copper solvent only injection is complete, approximately 400 gallons of hydrogen peroxide will be added (spiked), bringing the SG level up to 228" above the tubesheet (~12,100 gallons). The addition of the hydrogen peroxide will be immediately followed by air sparging at a rate of 650-1000 scfm per SG for ~15 minutes before proceeding to the next copper solvent injection.

After completing the sparging sequence, the remaining copper solvent (~18,000 gallons) will be injected into the SGs followed by another hydrogen peroxide injection of ~1,000 gallons. Additional hydrogen peroxide will again be added (spiked) to the SG. At this time, the SG levels will be at ~387" above the tubesheet (~31,000 gallons). Air sparging at 650-1000 scfm will be implemented for ~30 minutes following completion of the final hydrogen peroxide injection prior to collecting the first sample from SG#1 at full volume. Approximately 57 gallons of diammonium EDTA will be added to each SG with the refill sequence of the first sample sequence performed in each SG after reaching the final fill level ~387" above the tubesheet to achieve a desired total EDTA concentration of ~0.8 g/L.

Copper Step samples will be taken at various intervals starting at 2.0 hours after the start of the copper step and continuing throughout the entire 48 hour copper step. The final 48 hour drain down sample will be a composite sample from both steam generators collected at three different steam generator levels during draindown.

Both steam generators are then rinsed. This completes the chemical cleaning process. The entire steam generator chemical cleaning process is projected to take approximately five days.

See Section VI below for a complete description of the FANP equipment and plant connections required for the SGCC.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: 03-012	Sections I, II, III, IV, and VI required

Preparer: R.T. Finch / R. T. Finch / EOI / Design Engr. / 10-16-03
Name (print) / Signature / Company / Department / Date

Reviewer: R.L. Williams / R. L. Williams / EOI / Licensing / 10/16/03
Name (print) / Signature / Company / Department / Date

OSRC 10-16-03
Chairman's Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.9) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
1. Ron Stanley	1. SGCC "Process" Assistance.
2. Greg Hood	2. Environmental Review per Procedure EV-115.
3. T.C. Su	3. Equipment Qualification Assistance
4. Bob Varrin	4. Equipment Qualification Assistance

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☒	☐	No revision required.
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculation Manual[3]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☒ Yes ☐ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

[1] If "YES," see Section 5.1.4. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The plant systems which will be used for the proposed Steam Generator Chemical Cleaning are Blowdown, Controlled Ventilation Area, Condensate Makeup, Sump Pump, Main Steam and Station Air. These systems are described in the FSAR and drawings which show the specific details of the plant connection points are included in the FSAR by reference. Since the proposed SGCC has the potential to impact these plant systems Item 1 (FSAR) above was checked "Yes" and an Evaluation has been performed. Also, since the proposed SGCC is not described in the FSAR and Special Test Instruction has not been previously approved Item 2 above was checked "Yes".

Waterford 3 does not currently have an Independent Spent Fuel Storage Facility so Item 3 above was checked "N/A"

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LBDS_50_59	"steam generator chemical cleaning", "chemical cleaning", "blowdown", "CVAS", "Controlled Ventilation Area System", "Condensate Makeup", "Condensate", "Sump Pump", "Atmospheric Dump Valve", "Station Air"

LBDs/Documents reviewed manually:

UFSAR Sections 2.4.2.3, "Effects of Local Intense Precipitation"; 6.5.1.2.1.2, "Controlled Ventilation Area System"; 9.2.6, "Condensate Storage Facilities"; 10.3, "Main Steam Supply System"; 10.4.8, "Steam Generator Blowdown System"; 11.2, "Liquid Waste Management System". Technical Specification Section 3/4.7.7, "Controlled Ventilation Area System". 50.59 Report for ER-W3-2003-0366-005, "Evaluate Radiological and Environmental Controls During Construction Period of the Steam Generator Chemical Cleaning Process". 10CFR50.59 Evaluation No. 00-051, "50.59 Evaluation, Steam Generator Chemical Cleaning – Implementation". FANP Dwg. Nos. 6026845A, "Waterford 3 2003 Chemical Cleaning Sequence Control Procedure"; 6027358A, "Waterford 3 2003 Chemical Cleaning Process Control Procedure"; 51-5030944-00, "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report". Dominion Engineering, Inc. Letter No. L-4160-01-2, "Material and Geometry Review – Waterford 3 Steam Generator Chemical Cleaning", September 15, 2003.

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☒ Yes
☐ No

If "Yes," list the required changes.

1. 10CFR50.59 "Process" Exemption based on Evaluation No. 00-052.
2. 10CFR50.59 "Environmental" Exemption based on Evaluation No. 00-053.
3. Engineering Request No. ER-W3-2003-0366-02, Impact on Permanent Plant Instrumentation.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☒	☐	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☒	☐	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☒	☐	Involve the installation or modification of a stationary or mobile tank?
14.	☒	☐	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ☐ No

N/A

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

The Combustion Engineering 3410 Series steam generators (SGs) at Waterford 3 (W3) were chemically cleaned during the period of October 14 to October 19, 2000 by Framatome / Siemens. This secondary side cleaning was accomplished through application of the Siemens High Temperature Chemical Cleaning process. The process included both iron and copper removal steps. The W3 application was the first performance of the Siemens process in the United States.

The chemical cleaning and subsequent sludge lancing were successful in removing a significant amount of iron and deposit from the steam generators. A total of approximately 9,650 lbs of magnetite was removed from the SGs via chemical cleaning. This value includes both deposit dissolution and corrosion iron. Approximately 3,710 lbs of wet sludge was removed via sludge lancing following the chemical cleaning and another ~900 lbs (~300 lbs from SG #1 and ~600 lbs from SG #2) via sludge lancing following a post-cleaning upper bundle flush (UBF). Although a significant amount of deposits were removed during the 2000 chemical cleaning, residual deposits, specifically copper, need to be removed from the W3 SGs.

The proposed change is to implement a Framatome Advanced Nuclear Power, Inc. (FANP) low temperature (Mode 5) steam generator chemical cleaning during Refuel 12 to remove the residual steam generator deposits. An iron and copper removal step will be implemented to pursue optimal removal of targeted contaminants, copper and lead, within the allowable outage schedule. Removal of the copper is expected to improve the steam generator inspection eddy current signals and potentially improve W3's probability of detection (POD). In addition, the removal of the residual deposits may potentially correct the pressure loss across the steam generators.

The objectives of the cleaning are to remove the remaining available iron deposits present in the SGs with a low temperature (Mode 5) iron removal application to minimize the corrosion impact to the SG internals, and remove as much of the residual copper in the SGs as possible, especially in the crevice regions at the tube and eggcrate support intersection. The primary target for removal of the residual copper is based on a deposit characterization of ~3% in the deposit scale and ~10% in the loose powder deposit. Deposit amounts have been estimated at a nominal loading of 1,500-2,000 pounds of magnetite per steam generator with a maximum worst case loading of up to 3,000 lbs of magnetite per steam generator.

A summary of the steam generator chemical cleaning (SGCC) process is as follows:

1. Iron Steps (removes magnetite deposits)

   a. Performed when the Reactor Coolant System (RCS) is at approximately 190°F to 195°F (Mode 5).
   b. Both steam generators are drained to plant systems.
   c. Approximately 12,100 gallons of iron solvent, composed primarily of Diammonium Ethylenediaminetetraacetic Acid (EDTA) is injected (first injection) into each steam generator at a rate of approximately 150 gallons / minute (GPM) per steam generator. The flow rate can range from 65 GPM to 150 GPM per SG. The solution is injected from the FANP tank farm via the blowdown system (via blind flange connections downstream of valves BD-107A and BD-107B). The Atmosphere Dump Valves (ADVs) (MS-116A and MS-116B) are opened during the solvent injection to vent the steam generators.
   d. The SGs are sparged with nitrogen, supplied from the FANP tank farm area through the blowdown connections, for five minutes on sparge and ten minutes off sparge. The

sparge cycle continually repeats and ensures that complete mixing of the cleaning solution occurs in the steam generators.

e. Every several hours samples are drawn from the steam generators by draining approximately 2,000 of iron solvent to the FANP tank farm (the solvent is reinjected after sampling). Based upon the sample results the steam generators will be completely drained to the FANP tank farm approximately 4-8 hours after the initial iron solvent injection. The draining is accomplished by closing the ADVs and pressurizing the steam generators with nitrogen, supplied from tanks located adjacent to the FANP tank farm and injected through the blowdown connections, to a pressure of approximately 60 psig.

f. Approximately 31,000 gallons of iron solvent is injected (second injection) into the steam generators. The level will be approx. 387" above the steam generator tubesheet. As during the first injection, nitrogen venting will be continuous and sampling will occur periodically.

g. The sample results are again used to determine when the iron solvent will be completely drained from the steam generators.

h. Approximately 31,000 gallons of iron solvent is injected (third injection) into the steam generators. The level will be approx. 387" above the steam generator tubesheet. As during the first and second injections, nitrogen venting will be continuous and sampling will occur periodically.

i. The sample results are again used to determine when the iron solvent will be completely drained from the steam generators. If the sample results are satisfactory this is the completion of the iron steps.

2. Drying Step

a. The Reactor Coolant System temperature is reduced to approximately 150°F.

b. Air is sparged through each steam generator at a rate of 650-1000 scfm (per steam generator) to dry the steam generators prior to the copper step. The air is introduced into the steam generators, from FANP compressors located adjacent to the FANP tank farm, through the blowdown connections. The ADVs are open during the drying step.

3. Copper / Lead Step (removes copper and lead deposits)

a. Performed when the RCS is at a temperature of approximately 110°F (maximum allowable temperature of approximately 140°F).

b. Approximately 5,800 gallons of copper solvent (primarily composed of Ethylenediamine (EDA)) is injected into each steam generator at a rate of approximately 150 GPM. In parallel, 200 gallons of hydrogen peroxide is injected into each SG, at a rate of approximately 10 GPM. Injection is from the FANP tank farm through the blowdown connections.

c. Another approximately 5,600 gallons of copper solvent is injected into each SG. Following this, approximately 400 gallons of hydrogen peroxide is then injected into each SG.

d. Air is sparged through each steam generator at a rate of 650-1000 scfm for approximately 15 minutes.

e. Approximately 18,000 gallons of copper solvent is injected into each SG followed by another injection of approximately 1,000 gallons of hydrogen peroxide.

f. Air is sparged through each steam generator at a rate of 650-1000 scfm for approximately 30 minutes.

g. Every several hours samples are drawn from the steam generators by draining approximately 2,000 gallons of iron solvent to the FANP tank farm (the solvent is reinjected after sampling).

h. For the first sample/reinject sequence approximately 57 gallons of EDTA will be added to each steam generator during the reinjection.

i. After the first sample sequence, a sparging/pressurization/depressurization sequence will be performed in both SGs. The SGs will be sparged with air at a rate of 650-1000 scfm for approximately 1 hour, during which time the ADVs will be open for a short period and then closed in order to build the pressure in the SGs to approximately 45 psig. Once a pressure of 45 psig is gained, sparging will be secured and the SGs held at approximately 45 psig for 30 minutes. After 30 minutes, the ADVs will be opened and the SGs depressurized. Then the sparging sequence will begin again and continue throughout the duration of the step.

j. A drain and refill sequence may be performed at some time after 24 hours of exposure. For the refill approximately 31,000 gallons of copper solvent will be injected into each SG. Sparging and sampling will be performed following the refill.

k. When the sample results are satisfactory nitrogen is injected to drain the spent solvent solution from the steam generators.

3. Low Volume Rinse

   a. The steam generators and connecting lines are blown down with nitrogen to remove residual chemicals.

   b. Approximately 3,000 gallons of demineralized water is injected into each steam generator and then drained by injecting nitrogen. A minimum of two low volume rinses may be performed. The rinse water is sampled during draining.

4. Final Rinse

   a. Approximately 31,000 gallons of demineralized water is injected into each steam generator for a full volume rinse and then drained by injecting nitrogen. The rinse water is sampled during draining.

   b. The steam generators and lines are blown down with nitrogen.

The entire steam generator chemical cleaning process is estimated to take approximately five days and ten hours. It should be noted that the above described chemical cleaning process may be modified/adjusted slightly, during the cleaning process, according to the chemical samples that are taken and analyzed during the actual cleaning. Any changes will be approved, in writing, by both FANP and Entergy personnel.

Waterford 3 has developed a Special Test Instruction (STI-W3-2003-005) for the Steam Generator Chemical Cleaning (SGCC) Project. The title of the STI is "Waterford 3 Steam Generator Chemical Cleaning Special Test Instruction". The STI addresses Waterford 3 Operations Department responsibilities during the chemical cleaning. Waterford 3 Operations personnel (EOI) will maintain overall control of the plant during the cleaning and will operate all permanent Waterford 3 equipment. FANP personnel will operate all FANP supplied equipment and will control the flow of chemicals, air and nitrogen into the permanent plant systems. FANP personnel will work according to their own procedures, which have been reviewed and approved by EOI. Close and accurate communication between the EOI and FANP personnel is essential to the success of the cleaning project. Training, specific to the chemical cleaning will be conducted for the EOI Operations personnel that will be on duty during the chemical cleaning. FANP personnel will also participate in this training. This will ensure that personnel from both organizations are familiar with the procedures and interface requirements prior to the start of the actual steam generator cleaning.

The FANP procedure for controlling the SGCC process is Drawing No. 6026845A, "Waterford 3 2003 Chemical Cleaning Sequence Control Procedure". The set up of the FANP equipment for the SGCC project is shown on the following drawings:

1. Drawing No. 6026331E, "Waterford 3 2003 Chemical Cleaning Site Layout", 3 sheets.
2. Drawing No. 6026253E, "Waterford 3 2003 Chemical Cleaning P & ID", 3 sheets.

3. Drawing No. 6027153E, "Waterford 3 2003 Waste Processing P & ID", 3 sheets.

The majority of the FANP equipment is associated with liquid waste water storage and treatment. A bermed area approximately 132' x 153' has been constructed in the Plant Controlled Area west of the Service Building (outside of the Protected Area). The following equipment will be stored within the berm:

1. Two – 40,000 gallon iron process/waste tanks
2. Two – 40,000 gallon copper process/waste tanks
3. Five – 25,000 gallon waste storage tanks
4. Two – 20,000 gallon batch release tanks
5. One – 2,500 gallon hydrogen peroxide tank
6. One – Vent Trailer
7. Two – 500 gpm injection pumps.
8. Two – 500 gpm drain pumps
9. Two – Rain pumps
10. Two – 500 gpm tank recirculation pumps
11. One – 40' tall Evaporator for use during waste processing.
12. One – Evaporator Support Trailer containing a 1.6 MW Boiler.
13. Various pipes/hoses used to transfer water/chemicals/air/nitrogen between the FANP equipment/plant.

The following additional equipment will be placed adjacent to the bermed area:

1. One - 20' Cooling Tower Trailer
2. One - 45' Transformer Trailer
3. Two - 40' Parts Trailer
4. One - 40' Chemical Addition Trailer
5. One - 40' Administration Trailer
6. One – 40' Maneuvering Trailer
7. One – Chemical Addition Trailer –Utilized for transfer of all bulk chemicals into the various process tanks during solvent dilution and adjustment
8. One - 40' Crew Trailer
9. Five - 40' Hose Trailer
10. One - 40' Berm Trailer
11. One - 40' Radiation Protection Trailer
12. One – 40' Pipe Trailer
13. One – 40' Waste Processing Trailer
14. One – 20' office Equipment Trailer
15. Two – 20' Berm Parts Trailer
16. Four – Air Compressors
17. One – Nitrogen Trailer
18. Various pipes/hoses used to transfer water/chemicals/air/nitrogen between the FANP equipment/plant

The following FANP equipment will be placed inside the Waterford 3 Protected Area (PA):

1. Two – Drain Pumps
2. One – Vent Tank / Pump Skid
3. Various pipes/hoses used to transfer water/chemicals/air/nitrogen between the FANP equipment/plant

FANP Steam Generator Chemical Cleaning equipment has also been connected to Waterford 3 specific plant structures, systems and components (SSCs) in support of the chemical cleaning of the Waterford 3 Steam Generators. This work has been previously authorized by Procedure PMC-004-008 "Installation Procedure Steam Generator Chemical Cleaning Equipment Installation". The scope of the work includes

adding temporary isolation valves to the Blowdown and Controlled Ventilation Area (CVAS) systems, connecting hoses/piping downstream of manual isolation valves/caps on the Condensate Makeup (CMU) system, routing hoses to the Sump Pump (SP) system and Solidification Building, routing hoses/piping to the vendor supplied chemical cleaning equipment, placing humidity probes in the exit plume area of the Main Steam System Power Operated Atmospheric Dump Valves (ADVs), placing high velocity fans adjacent to the ADVs and connecting valves/hoses/piping to the Station Air System. Scaffolding will be erected in various areas to provide support for the FANP hoses/piping. Trenching will be required in some areas where the FANP hoses/piping cross roadways/security areas. Additional details of the work are provided below:

Blowdown System: To provide interface connections to the BD System for SG chemical cleaning, ER-W3-00-0042-03-00 installed 2" diameter blind flanged branch connections on blowdown lines 5BD4-3 and 5BD4-4, near the Blowdown Flash Tank. Procedure PMC-004-008 has authorized removal of the blind flanges on each line, and installation of temporary flange/pipe assemblies that include 2" diameter root valves. The two 2" lines have been routed a short distance and connect to the single 4" FANP process (injection/drain) line. The 4" process line has been routed through the floor plug opening at elevation -4, down to the -35 elevation and then through the unused wall sleeve at elevation -27 (see CVAS description below). The line has been run through the West side wet and dry cooling tower areas, over the Nuclear Island flood wall at elevation +30 (adjacent to the dry cooling towers), across the yard, trenched below the North side roadway and then run behind the Maintenance Support Building (MSB), below the security fence and out to the FANP tank farm/waste treatment area located in the Owner Controlled Area. Temporary hose/pipe connections will also be made to blowdown piping downstream of valve BD-1253 to enable the draining of SGCC chemicals from the Blowdown Flash Tank.

Condensate Makeup System: Demineralized water is required for initial functional testing, flushing the hoses and equipment during the chemical cleaning process and for waste processing, tank washing, and laboratory use prior to, during and following the chemical cleaning process. The water will be provided from a flanged 3" diameter hose connection installed on an existing 6" diameter blind flanged connection downstream of normally closed manual isolation valve CMU-1222. Valve CMU-1222 is a portable demineralizer hookup connection on the discharge of the Condensate Transfer Pump, and it is located near the Secondary Vacuum Degasifier. The 3" hose has been routed west under the security fence and then to the FANP tank farm/waste reduction area.

Controlled Ventilation Area System: The 4" FANP process line enters the Reactor Building (RB) Wing Area at the unused 6" wall sleeve at elevation -27 adjacent to the West Side Wet Cooling Towers. The wall sleeve is not part of any plant system and is normally blind flanged on both ends. FANP has provided temporary piping assemblies that consist of mating 6" flanges, piping, reducers and 4" root valves which have been installed on each side of the wall penetration.

Sump Pump System: A temporary flow path using 2" hose/piping has been created to allow discharge of treated waste water, rain water, cooling water, and eyewash/shower water, to Dry Cooling Tower (DCT) Sump No. 1, located in DCT A from the FANP Batch Release Tank (BRT). The hose has been routed from the FANP Waste Treatment Area to the +30.00 elevation Nuclear Island Floodwall where a pipe assembly has been installed on top of the flood wall. This pipe supports the hose and is anchored to the top of the flood wall to secure the hose. The hose is routed down the face of an interior wall that separates the individual cooling tower bays, and is attached to the catwalks adjacent to the cooling tower fans. The routing and securing of the hose to the catwalk structural steel and/or handrails ensures that the hose cannot fall or strike safety related cooling tower equipment located in the DCT Area. The hose is terminated at a convenient floor drain and anchored to secure the free end of the hose at the drain.

The 2" line will also have a tee junction adjacent to the dry cooling tower area. The second 2" line runs along the top of the flood wall at elevation +30, over the top of the West Side Access Building, across the top of the Tool Room to the Solidification Building where it is connected to the temporary demineralizer skid.

Main Steam System: FANP Humidity Probes will be placed in the exit plume area of the Power Operated Atmospheric Dump Valves (ADVs). This work will not be performed until the plant reaches Mode 5. High velocity fans will also be positioned, adjacent to the ADVs, to accelerate the fumes that are released during the SGCC past the potential downdraft effects of the containment building.

Station Air System: One inch diameter FANP valves/piping/hoses have been connected to Station Air valves SA-712 and SA-8016. The connection to SA-712 provides air to power the FANP transfer pump located in the Reactor Building -35 elevation and the connection to SA-8016 provides air to power the FANP pump in the protected area outside of the Nuclear Island Floodwall. An air hose has been run over the + 30 elevation Nuclear Island Floodwall to supply air to the pump in the protected area. Except for the initial testing station air will only be used in Mode 5.

This 10 CFR 50.59 Safety Evaluation is for the implementation portion of the Steam Generator Chemical Cleaning (SGCC) only (evaluates Waterford 3 Steam Generator Chemical Cleaning Special Test Instruction - STI-W3-2003-005). It addresses mainly the effect of the cleaning process on Waterford 3 Technical Specifications and Updated Final Safety Analysis Report Chapter 15 accident analyses. The connection of the FANP chemical cleaning equipment to plant systems was evaluated separately as part of Procedure PMC-004-008 "Installation Procedure Steam Generator Chemical Cleaning Equipment Installation". ER-W3-2003-0366-02 addresses the impact of the Steam Generator Chemical Cleaning Process on Waterford 3 Permanent Plant Instrumentation. ER-W3-2003-0522-00 addresses the impact of SGCC chemicals on Fuel and Reactor Coolant System Components due to leakage [through steam generator tube(s)] from the secondary to primary side during the chemical cleaning. The effects on the steam generator and the environmental effects are addressed in separate 10 CFR 50.59 Reports; a "Process" Exemption and an "Environmental" Exemption. The separate reports address the following issues:

1. Chemical Process (see "Process" Exemption).
3. Control Room habitability (see "Environmental" Exemption).
4. Tank Farm Operation and non-FSAR Chapter 15 events (tornado, flooding and hurricane - see "Environmental" Exemption).
5. Decomposition and process by products (chemical fume impact on plant personnel - see "Process" and "Environmental" Exemptions).
6. Vendor equipment failure modes and effects analysis (see "Environmental" Exemption).
7. Test Results of Waterford 3 specific laboratory testing (see "Process" Exemption).
8. Process Corrosion monitoring (see "Process" Exemption).
9. Eddy Current tube testing following SGCC (see "Process" Exemption).
10. Waste Processing (see "Environmental" Exemption).
11. Environmental releases (see "Environmental" Exemption).

Reason for proposed Change:

The Combustion Engineering 3410 Series steam generators at Waterford 3 (W3) were chemically cleaned during the period of October 14 to October 19, 2000 by Framatome ANP, Inc. This secondary side cleaning was accomplished through application of the Siemens High Temperature Chemical Cleaning process. The process included both iron and copper removal steps. The W3 application was the first performance of the Siemens process in the United States.

The chemical cleaning and subsequent sludge lancing were successful in removing a significant amount of iron and deposit from the steam generators (SGs). A total of approximately 9,650 lbs of magnetite were removed from the SGs via chemical cleaning. This value includes both deposit dissolution and corrosion iron. Approximately 3,710 lbs of wet sludge were removed via sludge lancing following the chemical cleaning and another ~900 lbs (~300 lbs from SG #1 and ~600 lbs from SG #2) via sludge lancing following a post-cleaning upper bundle flush (UBF). Although a significant amount of deposits were removed during the 2000 chemical cleaning, residual deposits, specifically copper, need to be removed from the W3 SGs. Removal of the copper is expected to improve the steam generator tube inspection eddy current signals and potentially improve the probability of detection (POD) of tube flaws. In addition, the removal of the residual deposits may potentially correct the pressure loss across the steam generators.

The objectives of the cleaning are to remove the remaining available iron deposits present in the SGs and remove as much of the residual copper in the SGs as possible, especially in the crevice regions at the tube and eggcrate support intersection. The primary target for removal of the residual copper is based on a deposit characterization of ~3% in the deposit scale and ~10% in the loose powder deposit. Deposit amounts have been estimated at a nominal loading of 1,500-2,000 pounds of magnetite per steam generator with a maximum worst case loading of up to 3,000 lbs of magnetite per steam generator.

50.59 Evaluation summary and conclusions

The proposed change does not cause the parent systems to be operated outside of their design or test limits, negatively affect any system interfaces or result in an increase in challenges to safety systems or systems important to safety. The proposed activity does not result in a change from one frequency class to a more frequent class or an increase in frequency within a given class. This evaluation concludes that neither the actual chemical cleaning, or the temporary system connections associated with the chemical cleaning, will degrade the integrity or performance of the steam generators, the connected instrumentation, or the affected systems. All physical changes are temporary, and there are no new permanent system interactions created. This change does not require any Technical Specification changes. An Environmental Impact Evaluation was required. As stated above the Process effects internal to the steam generator and the Environmental effects of the proposed Steam Generator Chemical Cleaning are addressed in separate 10 CFR 50.59 Reports (see the 10 CFR 50.59 "Process" Exemption and 10 CFR 50.59 "Environmental" Exemption).

B. License Amendment Determination

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 FSAR? ☐ Yes ☒ No

BASIS:

The probability of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR) will not be increased due to the implementation of the Steam Generator Chemical Cleaning process. The probability of a rupture of a steam generator tube should actually be decreased by the SGCC due to the removal of deposits that can induce corrosion of the steam generator tubes, tubesheets and other internal components. The purpose of the cleaning process is to enable the steam generators to reach their forty-year design life. The chemical cleaning process and the implementation of this process are evaluated below with respect to the Waterford 3 Technical Specifications and to the Updated Final Safety Analysis Report Chapter 15 analyses. To fully answer this question the Technical Specification compliance issues will be addressed first, then the UFSAR Chapter 15 analyses will be addressed.

During implementation of the Steam Generator Chemical Cleaning (SGCC) process Waterford 3 will be operated in accordance with Technical Specifications (TS) Sections 3.4.1.4, 3/4.7, 3/4.11 and 6.0. The iron and copper/lead steps will be performed in Mode 5 (Cold Shutdown). Mode 5 is defined as K_{eff} less than .99 and RCS temperature less than or equal to 200°F. During the SGCC iron step the RCS temperature will be maintained between approximately 190°F and 195°F. During the SGCC drying step the RCS temperature will be maintained at approximately 150°F. During the SGCC copper/lead step the RCS temperature will be maintained between approximately 110°F and 140°F. The RCS conditions for the SGCC iron step, drying step and copper / lead step are thus bounded by Mode 5 requirements. Some of the Technical Specifications discussed below, such as 3.4.4, 3.6.6.1 and 3.7.7 are applicable only in Modes 1, 2, 3 and 4 and so do not apply during the Mode 5 SGCC. They are addressed herein because there is the potential that the SGCC could result in changes that would impact the plant when power operation is resumed during Cycle 13.

Technical Specification 3.4.1.4 requires that at least two of the loops(s)/trains listed below shall be Operable and at least one reactor coolant and/or shutdown cooling loop shall be in operation in Mode 5:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump.

3. Shutdown Cooling Train A.
4. Shutdown Cooling Train B.

Technical Specification Surveillance Requirement 4.4.1.4.2 requires that the required steam generator shall be determined Operable by verifying the secondary side water level to be ≥ 50 % of wide range indication at least once every twelve hours. For the SGCC iron step, drying step and copper/lead step, the steam generators are not operable because the water levels will be below 50 %. This is acceptable because both trains of shutdown cooling will be Operable and at least one train will be in operation.

UFSAR Section 9.3.6.3.4 discusses a Loss of Shutdown Cooling with the Reactor Coolant System Partially Filled. The UFSAR section was in response to Nuclear Regulatory Commission Generic Letter 87-12 dated July 9, 1987. In the event that shutdown cooling is lost during the time that the steam generators are inoperable, during SGCC, their operability can be restored by terminating the chemical cleaning and refilling the steam generators with *auxiliary feedwater.*

Technical Specification 3.7.2 requires that the temperature of the secondary coolant in the steam generators shall be greater than 115°F when the pressure of the secondary coolant is greater than 210 psig. The copper/lead step will be performed at approximately 110°F at a maximum steam generator secondary side pressure of 70 psig (Reference STI-W3-2003-0005-00). These parameters are within the requirements of the TS 3.7.2.

Technical Specification 3.4.4 requires that each steam generator shall be Operable in Modes 1, 2, 3 and 4. Operability is determined during shutdown by selecting and inspecting the steam generator tubes per Technical Specification Table 4.4-2. The purpose of the SGCC is to facilitate the steam generator tube inspection by removing the copper that may potentially interfere with the (eddy current) inspection. Following the proposed steam generator chemical cleaning, the steam generator tubes will be inspected per the Technical Specification requirements.

Technical Specification 3.11.1.4 sets limits on the quantity of radioactive material that will be stored in outside temporary tanks. The temporary storage and disposal of radioactive liquid waste water generated during chemical cleaning is evaluated in one of the separate 10 CFR 50.59 evaluations mentioned above (see 10CFR50.59 "Environmental" Exemption). The *conclusion of the "Environmental" Exemption is* that the liquid waste generated by the SGCC will be in compliance with this Technical Specification.

It should be noted, as previously stated, that the amount of toxic gases or fumes to be released should not present a personnel safety risk. Small quantities of ammonia and hydrazine will be released, but not in sufficient quantities to present a hazard. Calculations have been performed to determine the concentration and dispersion of the fumes. The calculation results show that the fume concentration is within all of the acceptable

regulatory and toxicity limits. See the separate 10 CFR 50.59 "Environmental" Exemption for a complete discussion on this subject.

Monitoring will also be conducted to determine the concentration of chemical fumes at key locations in and around Waterford 3 (see the separate 10 CFR 50.59 "Environmental" Exemption for a thorough discussion of this issue). Both fixed and mobile chemical detectors may be utilized for the monitoring. This monitoring will help to protect plant personnel from any adverse effects due to chemical release.

Information obtained from Nuclear Regulatory Commission (NRC) Information Notice (IN) 95-41, "Degradation of Ventilation System Charcoal Resulting From Chemical Cleaning Of Steam Generators" will be utilized during the Waterford 3 SGCC project to maintain compliance with the Waterford 3 Technical Specifications. The pertinent facts from this information notice are as follows:

1. In June 1994 the Surry Unit 2 Steam Generators were chemically cleaned using basically the same chemicals that will be used to clean the Waterford 3 steam generators (EDTA, EDA and hydrazine etc.).
2. Following the SGCC the steam generators were opened to allow access for sludge lancing. Concentrations of ammonia and hydrazine, in containment, were then measured at 30 ppm and 6 ppm respectively.
3. Both train A and train B of the auxiliary ventilation system were run, for approximately eight hours to reduce the chemical concentrations in containment.
4. The charcoal in both trains was then sampled via laboratory tests for the methyl iodide removal efficiency. The removal efficiency for trains A and B was 93.4 % and 90.7 % respectively. These efficiencies were below the Technical Specification requirements.

The conclusion in the Information Notice is that discharging the air involved in steam generator chemical cleaning operations through systems containing charcoal was likely to degrade the charcoal. The degradation mechanism was thought, by the NRC, to be breakdown of the hydrazine into water (along with EDA and EDTA as contributors). The water then degraded the methyl iodide removal efficiency of the charcoal.

During SGCC the Control Room Ventilation system will be placed in the "Isolate" mode. This will ensure the operability of the system by isolating the charcoal adsorber in the system from contamination and degradation due to contact with chemicals released during the SGCC. Technical Specification 3/4.7.6.1 is not applicable because this TS is applicable only in Modes 1, 2, 3 and 4. Note that there will be times during the SGCC when no chemical fumes will be released. During these times operation of the system may be resumed at the discretion of the Operations Department. This will enable the carbon dioxide levels in the control room to be maintained at acceptable levels.

Shield Building Ventilation (SBV) is to remain secured in Mode 5 when the Shield Building Hatch is closed. When in Mode 5 with the hatch open the Process Lead must verify that ammonia and hydrazine vapors are less than

one part per million (PPM) at the hatch opening before starting SBV. If the vapors are equal to or exceed 1 PPM then SBV start will be delayed until System Engineering recommends a course of action. If SBV is in operation and vapors are equal to or exceed 1 PPM then SBV will be secured and System Engineering will be notified to evaluate the need to sample the filter charcoal. This will ensure compliance with Technical Specification 3/4.6.6.1 even though this TS is only applicable in Modes 1, 2, 3 and 4. If a radiological release occurs the system will be allowed to run normally, whether or not there are chemical fumes present. This is shown to be acceptable per the following discussion:

The Loss of Coolant Accident (LOCA) and Non-LOCA scenarios must be considered with respect to charcoal filter degradation affects on event consequences. For a lower mode LOCA (Mode 4 or Mode 5) Combustion Engineering provided an Infobulletin on Emergency Core Cooling System (ECCS) requirements needed to mitigate the consequences of a LOCA (ABBCE Infobulletin No. 99-01). This Infobulletin stated that with one High Pressure Safety Injection (HPSI) train available within 10 minutes, a severance of the largest line connected to the RCS will not uncover the core or increase the fuel cladding temperature. This means that the typical design basis LOCA radiological source terms will not be present and the safety charcoal filters will not be needed to meet the typical LOCA design basis requirements. Thus the issue of potential charcoal filter degradation is not predicted to produce adverse consequences for the Mode 4 and 5 LOCA due to no predicted fuel failure.

For the lower mode Non-LOCA events, the expected RCS and SG activities (RCS < 0.01 uCi/gm as of 9/27/00) are a factor of 100 below those used in the accident analyses. Non-LOCA events initiated from lower mode operation are not predicted to incur fuel failure. Thus it is expected that potential charcoal filter degradation would have no affect on the accident consequences.

During SGCC the Controlled Ventilation Area System Filter Trains will be secured. Technical Specification 3/4.7.7 is only applicable in Modes 1, 2, 3 and 4. If the CVAS Filter Train must be placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV system. Also as described above the system will be placed in service in the event of a radiological release regardless of chemical vapors. Chemical monitoring will be conducted for chemicals at the outside air intake for the RAB Normal Supply System and the Reactor Auxiliary Building (RAB) – 4 wing area. Sampling of the charcoal in the Controlled Ventilation Area System (CVAS) will also be conducted after the completion of the SGCC if monitoring results indicate that sampling is required. Also as stated above chemical fume degradation of the charcoal would have no affect on the accident consequences for a Mode 4 or Mode 5 accident (SGCC is performed in Mode 5).

During SGCC the Fuel Handling Building (FHB) Emergency Ventilation and Normal Ventilation will be secured. If the FHB Emergency Ventilation must be

placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV system.

For all of the above mentioned safety related, charcoal containing, HVAC systems the monitoring during SGCC in conjunction with potential (if chemical fumes are detected) charcoal sampling will ensure, and demonstrate compliance with the applicable Technical Specifications when full power operation is resumed following RF 12 (the SGCC will be performed in Mode 5 when the operation of the HVAC systems is not required by Technical Specification. If charcoal testing is warranted, due to fumes released during SGCC, and the testing reveals that any of the safety related charcoal adsorber beds is below the Technical Specification efficiency, the charcoal will be replaced prior to resuming full power operation. This ensures compliance with the applicable Technical Specifications after SGCC.

It should be noted that dedicated communication lines (telephones and/or radios) will be established linking the EOI Waterford Maneuvering Representative (definition from FANP Procedure Dwg. 6026845A) with the Control Room. The EOI Maneuvering Representative will be responsible for coordination of all activities. In addition, the Maneuvering Representative will be the single Waterford 3 point of contact for the evolution. This helps to minimize any risks from the chemical cleaning activity by providing immediate notification to the Waterford 3 Control Room of any potential problem.

The UFSAR accident analyses clearly bound the chemical cleaning evolution. The plant conditions that will be maintained during this evolution are RCS temperature maintained at a maximum of approximately 197°F, RCS pressure in accordance with the RCP operating curves and steam generator pressure and temperature at saturation conditions. Note that the saturation pressure corresponding to a temperature of 200 °F (upper limit of Mode 5) is approximately 11.5 psia. The stable plant conditions that are maintained during this process make it highly unlikely that a plant event would occur as a result of the chemical cleaning process. Comparison of these plant conditions to the conditions assumed in the UFSAR analyses demonstrates that the UFSAR analyses represent worst case scenarios and clearly bound the steam generator chemical cleaning evolution. No aspect of the chemical cleaning activity increases the probability of an accident that was previously analyzed in the UFSAR.

The Blowdown System, Condensate Makeup System, and Sump Pump System are not postulated to initiate any Chapter 15 accidents. Chapter 15 initiating events that have been identified as having the potential to be affected by the SGCC are as follows:

1. Increased Main Steam Flow (UFSAR Section 15.1.1.3)
2. Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (UFSAR Sections 15.1.1.4 and 15.1.2.4)
3. Steam System Piping Failures (UFSAR Sections 15.1.3.1, 15.1.3.2, 15.1.3.3)
4. Loss of Normal AC Power (UFSAR Section 15.2.1.4)

5. Primary Sample or Instrument Line Break (UFSAR Section 15.6.3.1)
6. Steam Generator Tube Rupture (UFSAR Section 15.6.3.2)
7. Liquid Waste System Leak or Failure (Release to Atmosphere – UFSAR Section 15.7.3.2)
8. Postulated Radioactive Releases Due to Liquid Containing Tank Failure (UFSAR 15.7.3.3)

Items 1 and 2: Increased main steam flow (item No. 1 above) can be caused by the inadvertent opening of an atmospheric dump valve (or safety valve). The failure scenario would be that the release of the chemical cleaning vapors through the ADVs during SGCC would damage the ADV (or safety valve) to such an extent that the ADV would fail to seat properly either during the SGCC or following RF 12 (during cycle 13 or following cycles). The vapors produced during the iron step will be composed primarily of water and ammonia with a small amount of hydrazine and other chemical cleaning chemicals (carryover from the steam generator) also present. The copper step vapors will be essentially all water and ammonia entrained in air with small amounts of the chemical cleaning chemicals (carryover from the steam generator). Materials evaluations, presented in FANP Document No. 51-5030944, show that this is not a credible failure mechanism. Similar justification can be used to rule out the inadvertent opening, due to component failure, of a Steam Generator Atmospheric Dump Valve (item No 2 above). As added justification of the acceptability of the SGCC process it should also be noted that there have been no SGCC induced failures of valves or piping at other plants that have cleaned their steam generators. Previous inspections, at other plants, have also not identified any unexpected (expected amounts of metal loss range from negligible for metal exposed to steam and gaseous cleaning chemicals to small amounts, within the design allowances, for metal exposed to the chemical cleaning liquids) material degradation following SGCC.

Item 3: From a purely engineering standpoint a steam System Piping failure is not very likely to occur during the Mode 5 SGCC because of the low temperatures and pressures associated with the SGCC. These piping breaks are postulated as part of the plant licensing basis in other modes (i.e. UFSAR Section 15.1.3.2 postulates steam system piping failures inside and outside containment in Mode 3 and Mode 4) and during the next operating cycle pressures and temperatures will be at higher operating levels i.e. nominal pressures and temperatures.

The SGCC iron steps, drying step and copper/lead steps will occur during Mode 5. The Technical Specifications define the Mode 5 average RCS coolant temperature as less than or equal to 200 °F. As stated above the saturation pressure at a temperature of 200 °F is approximately 11.5 psia. The maximum pressure in the steam lines during SGCC will therefore not exceed the pressure imparted from the FANP process (solution injection and draining and sparging). The upper limit on the SGCC process pressure is 70 psig (Reference STI-W3-2003-0005-00). The design pressure and temperature for the main steam lines is 1085 psig and 555 °F respectively.

The maximum operating pressure and temperature is 985 psig and 545 °F respectively. Immediately prior to the SGCC the main steam system will have been subjected to operating pressures and temperatures below the design conditions but much above the conditions that will be present during the cleaning. It is therefore extremely unlikely that the steam lines would fail at the SGCC pressure and temperature of 70 psig and 200 °F respectively.

The probability of a Steam System Piping Failure is not increased due to SGCC because the chemicals (in both liquid and vapor form) introduced into the piping during the SGCC will not reduce the strength of the piping. As stated above the main steam lines will have mostly water (as steam), ammonia and slight amounts of hydrazine and other chemical cleaning chemicals in them during SGCC. Materials evaluations, presented in FANP Document No. 51-5030944, show that there will be no detrimental effect on the steam system piping or valves.

Although a break in the steam generator blowdown piping is not an accident previously evaluated in the UFSAR, to completely evaluate the issue of potential pipe breaks this issue will also be addressed in the response to this Question. Additional FANP documents show that there will be no detrimental effect, due to corrosion and or erosion, on the Waterford 3 blowdown piping due to SGCC. The titles and identification numbers of these documents are as follows:

1. "Waterford Unit 3 Material and Geometry Review and Evaluation" (FANP Document Identifier 5008453-00).
2. "PVNGS – 2 Chemical Cleaning Materials Evaluation / Corrosion Estimates" (FANP Document Identifier 51-1228499-00, referenced in FANP Document 5008453-00).
3. "PV-1 HTCC Material Evaluation" (FANP Document Identifier 51-1234952-00, referenced in FANP Document 5008453-00).
4. "Salem Unit 2 Materials Evaluation" (FANP Document Identifier 51-1269270-00, referenced in FANP Document 5008453-00).
5. "300 F Chemical Cleaning Velocity Test Report" (FANP Document Identifier 51-1234107-00, referenced in FANP Document 51-1234952-00).
6. "PVNGS-1 High Temperature Chemical Cleaning Research Qualification Test" (FANP Document Identifier 51-1234937-00, referenced in FANP Document 51-1269270-01).
7. "PV-1 High Temperature Chemical Cleaning Optimization Test Report" (FANP Document Identifier 51-1234949-00, referenced in FANP Document 51-1269270-01).
8. Dominion Engineering, Inc. Report R-4135-00-1, Rev. 0, "Summary of Nominal Process and Low Deposit Loading Autoclave Tests for Qualification of the Siemens Chemical Cleaning Process at Waterford 3", August 2000.
9. "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report", (FANP Document Identifier 51-5030944-00), September 29,2003.
10. Dominion Engineering,Inc. Letter L-4160-01-2, "Material and Geometry Review – Waterford 3 Steam Generator Chemical Cleaning", September

15,2003.

The reports demonstrate that the corrosion/erosion of the SA-106 Grade B blowdown piping at Waterford 3, during SGCC, will be well within the allowable design limits. There is therefore no potential for the proposed SGCC to cause a break of the blowdown piping or blowdown nozzle.

The cleaning chemicals that will be injected into the steam generators through the blowdown piping will be in liquid form at approximately ambient atmospheric temperature (less than 100 °F). It should be noted that at temperatures below approximately 250 °F the metal/chemical reaction rate of the chemicals used in the cleaning process is very low. This low reaction rate, at low temperature, minimizes the impact that the chemicals have on the blowdown piping as they are injected into the steam generators.

The temporary FANP supplied lines, hoses and pumps do not create the possibility of a new High Energy Line break. The pressure and or temperature operating conditions of the temporary lines and components are below the HEL break threshold.

Item 4: Any SGCC affect on the probability of a Loss of Normal AC Power accident (item No. 4 above) can also be eliminated. The power supply for the vendor supplied pumps and major equipment is taken from outside of the Waterford 3 plant (an independent transformer is connected to the Entergy electrical grid outside of the plant protected area with power cables run into the protected area through the security zone). Internal plant electrical power is supplied only to limited vendor portable equipment from existing 110 volt receptacles. The 110-volt receptacles are not capable of initiating any UFSAR Chapter 15 accidents.

Item 5: Effects on the probability of a Primary Sample or Instrument Line Break (item No. 5 above) can also be eliminated. The stainless steel tubing and piping that is used on the primary system is unaffected by the chemicals used in the proposed SGCC process. The chemicals would therefore be incapable of causing a break even if they were to enter the primary system. It should be noted here that ER-W3-2003-0366-02 provides guidance for isolating some of the permanent plant instrumentation that has the potential to be affected by SGCC. This instrumentation is all on the secondary system. The isolated instrumentation will therefore not be impacted by the SGCC. For the permanent plant instrumentation that has the potential to be impacted but that is not isolated during SGCC, the ER also justifies that there will be no harm done to this instrumentation. See the ER for further information on this subject.

Item 6: The separate "Process" 10 CFR 50.59 Report addresses the effects of the SGCC on the steam generator tubes and steam generator Internals (item No. 6 above).

Items 7 and 8: The separate "Environmental" 10 CFR 50.59 Report also evaluates the effect of SGCC on the potential liquid releases (items No. 7 and 8 above).

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 FSAR? ☐ Yes ☒ No

BASIS:

The probability of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. As stated above, the impact of the chemicals used in the SGCC on the Inconel 600 tubes that provide the barrier for fission product release will be addressed in a separate 10CFR 50.59 Report. There will be however no adverse impact on the steam generator tubes. Corrosion of the steam generators and affected plant systems materials will be well within the design corrosion allowances. Foreign material control will be aggressively pursued in accordance with existing plant administrative control procedures. All of the FANP supplied equipment, hoses etc. will be flushed and tested prior to use. The temporary equipment will be operated and the discharges will be sampled (in some cases with the use of filter bags) until the equipment is proven to be clear of any foreign material (dust, sand, scale etc). This ensures that no foreign material or loose parts, which could conceivably cause a tube failure during plant operation, would be inadvertently injected into the steam generator or other plant systems. The proposed steam generator chemical cleaning will not require that any plant structures, systems or components operate outside of their design bases.

The effect of the release of evaporated cleaning chemicals through the ADVs has been evaluated in FANP Document 51-5030944, "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report". That evaluation confirms that the constituents of the vapor that will be released through the ADV will have no adverse impact on the ADV materials. Therefore, failure of the ADV, as a result of SGCC, is not a credible event.

Temporary system connections to permanent plant equipment are easily isolated or disconnected in the event of an emergency. No temporary system connections are made within the Containment building. The location of all temporary system connections and the required chemical cleaning equipment have been chosen to minimize the potential impact on equipment important to safety.

The SGCC process/injection lines/hoses will carry the chemical cleaning chemicals, ammonia and hydrogen peroxide. These chemicals could have the potential to increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR if the new chemicals could cause unanalyzed damage to safety related components that they may come in contact with during a potential process line/hose break. The following addresses this issue:

Effect of Chemical Cleaning Operations and Potential Impact of Off-Normal Conditions on Integrity and Operability of Safety Related Structures, Systems and Components at W3

The purpose of this evaluation is to assess the impact of RF12 steam

generator chemical cleaning operations on safety related equipment in the event that this equipment is exposed to liquid or gaseous species during normal or off-normal cleaning operations. Normal operations would include emissions associated with sparging air or nitrogen through the steam generators, and filling and draining of the chemical cleaning system including hoses, lines and tanks. Off-normal conditions would range from unanticipated fugitive emissions to liquid or gaseous exposures due to leaks or spills.

Prior to the chemical cleaning in RF10 (2000), a similar evaluation of the environmental qualification (EQ) of safety related equipment was conducted. In this evaluation, it was determined that normal and off-normal emissions would not represent a risk to EQ equipment. This was due to the fact that: (1) the two solvents used in RF10 (iron and copper solvent) had pHs that were within the bounds of that previously determined to be acceptable per the EQ program (4.5 to 10), and (2) air emissions from SGCC operations were too low to represent a corrosion risk to equipment.

For the chemical cleaning in RF12, iron and copper solvents will again be used. The iron solvent is still EDTA-based but applied at lower temperature, 190 to 195°F as opposed to 320 to 345°F, and without "venting" through the ADVs. Nitrogen sparging will be used instead of venting to promote mixing in the steam generators. The pH of the iron solvent will be 6.9 to 7.8 during preparation, and will be limited to 6.9 to 9.0 during application as described in FANP Document 6027358, "Waterford 3 2003 Chemical Cleaning Process Control Procedure". Accordingly, the pH is within the bounds of the EQ program (less than 10.5 but greater than 4.5). The emissions of hydrazine, and ammonia associated with sparging in lieu of venting are similar to those associated with the RF10 cleaning as documented in FANP Document 51-5030944 "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report" and FANP Document 51-5031596-01 "Waterford Unit 3 2003 Chemical Cleaning Emissions Estimates". As such, the SGCC iron step application is again considered acceptable with regard to exposure of safety related equipment to SGCC emissions or spills and leaks.

The copper solvent that will be used in RF12 is also similar to that used in RF10, a mixture of EDA and ammonium carbonate, with a small amount of EDTA. The only significant change to the solvent is that 20 g/l hydrogen peroxide (about 2 weight percent) will be added early in the process to improve copper dissolution kinetics. Emissions estimates performed by FANP show that the release of the only species with significant volatility, other than ammonia which was considered in RF10, in the copper step (EDA) due to air sparging is small and similar to that evaluated in RF10. The volatility of hydrogen peroxide is 300 times lower than that of EDA or ammonia, and its expected concentration in air either at the ADVs or adjacent to spills or leaks is very low (calculated to be only 3 ppm). At this concentration, peroxide is essentially non corrosive to ferritic and austenitic steels, and will not corrode typical materials of construction used

in instrumentation and controls (elastomers, plastics, rubber, Viton, etc.). As such, the SGCC copper step application is again considered acceptable with regard to exposure of safety related equipment to normal SGCC emissions.

Auxiliary Component Cooling Water (ACCW) Pump A is immediately adjacent to the inside of the wall penetration (elevation -27) through which the 4" FANP process line enters the Reactor Building. If the process line were to break and spray chemicals directly onto the pump motor, and if the chemicals were to enter the motor and contact the copper components in the motor, the motor would be severely damaged or destroyed. To protect the motor a scaffold barrier will be erected over the ACCW pump to prevent chemicals from entering the motor. The scaffold barrier will be erected in accordance with procedure PMC-002-006, "Erecting Scaffolding" and Special Test Instruction, STI-W3-2003-0005-00, "Special Test Instruction Waterford 3 Steam Generator Chemical Cleaning". The materials used to erect the scaffold barrier will either be chosen for their resistance to the SGCC chemicals or protected from degradation by the SGCC chemicals. The scaffold barrier will be constructed so as not to obstruct cooling air flow to the pump/motor.

In addition, while the nominal pH of the copper solvent is 10 as it was in RF10, FANP Document 6027358 "Waterford 3 2003 Chemical Cleaning Process Control Procedure" sets the allowable range at 9.5 to 12. This pH range in excess of 10.5 is outside the acceptable range of the EQ program of 4.5 to 10. The pH in excess of 10 is due to use of higher concentrations of EDA in the solvent to maximize solvent capacity for copper dissolution. Accordingly, a review of the corrosiveness of this additional EDA on typical safety related materials of construction was completed in preparation for the chemical cleaning in RF12. Corrosion rates for nine different materials due to exposure to EDA were obtained (silicon rubber, epoxies, polyolefins, natural rubber, carbon and low alloy steel, stainless steel, galvanized steel, heat shrink tubing, EPR and Viton). Corrosion rates for all non steel materials were essentially zero. Very slight corrosion to carbon and stainless steel was reported in the literature (less than 0.01 mils per hour). This is considered negligible, and as such exposing safety related equipment to a copper solvent as a result of a spill is considered acceptable. If a spill does occur a Condition Report will be initiated and the impacted equipment will be inspected and evaluated.

The only safety related portion of Blowdown is between BD-103A(B), (Blowdown Outside Containment Isolation Valve) and the steam generators. The proposed change does not affect the safety related portion of the blowdown system. The cleaning process design will limit corrosion of steam generator secondary side components and BD base material to within corrosion allowances. The proposed Special Test Instruction directs draining the steam generators using blowdown to Circulating Water (CW) or the Condenser Hotwell. In the case of the condenser hotwell, the BD system will be operated in accordance with normal operating procedures. These portions of the Blowdown system are not safety related.

The proposed Special Test Instruction provides instructions for sparging the steam generators with compressed air. The normal operating pressure of the steam generators bounds the 70 psig applied during sparging. Thus, the probability of a malfunction of the steam generators is unaffected by the proposed air sparging.

Nuclear Regulatory Commission (NRC) Information Notice (IN) 95-41 describes an event in which cleaning chemicals damaged Heating, Ventilation and Air Conditioning (HVAC) Systems equipped with charcoal filters. Since the Special Test Instruction may introduce cleaning chemicals into areas ventilated by systems containing charcoal filters, the probability of a malfunction of HVAC systems equipped with charcoal filters is potentially affected.

In order to protect the safety related charcoal in the plant Heating, Ventilation and Air Conditioning systems the following actions will be taken during the proposed Waterford 3 SGCC:

1. The Control Room Ventilation system will be placed in the "Isolate" mode.
2. Shield Building Ventilation (SBV) is to remain secured in Mode 5 when the Shield Building Hatch is closed. When in Mode 5 with the hatch open the Process Lead must verify that ammonia and hydrazine vapors are less than one part per million (PPM) at the hatch opening before starting SBV. If the vapors are equal to or exceed 1 PPM then SBV start will be delayed until System Engineering recommends a course of action. If SBV is in operation and vapors are equal to or exceed 1 PPM then SBV will be secured and System Engineering will be notified to evaluate the need to sample the filter charcoal.
3. The Controlled Ventilation Area System (CVAS) will be secured. If CVAS must be placed in service then the Process Lead must verify that the ammonia and hydrazine vapors are less than 1 PPM prior to starting CVAS. If the vapors are equal to or exceed 1 PPM then CVAS start will be delayed until System Engineering recommends a course of action. If CVAS is in operation and vapors are equal to or exceed 1 PPM then CVAS will be secured and System Engineering will be notified to evaluate the need to sample the filter charcoal.
4. During SGCC the Fuel Handling Building (FHB) Emergency Ventilation and Normal Ventilation will be secured. If the FHB Emergency Ventilation must be placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV and CVAS systems.

Note that there will be times during the SGCC when there will be no likelihood of chemical fumes being released. During these times the operation of the various ventilation systems may be restored at the discretion of the Operations Department. It should also be noted, as previously stated, that the amount of toxic gases or fumes to be released should not present a personnel safety risk. Small quantities of ammonia, hydrazine and other cleaning chemicals (EDTA and EDA) will be released, but not in sufficient quantities to present a hazard. Monitoring will be conducted to determine the

concentration of chemical fumes at key locations. Both fixed and mobile monitors may be utilized. This monitoring will help to protect plant personnel from any adverse effects due to chemical release.

An increase in the probability of a Primary Instrument Line Break can also be eliminated. The stainless steel tubing and piping that is used on the primary system is unaffected by the chemicals used in the proposed SGCC process. The chemicals would therefore be incapable of causing a break even if they were to enter the primary system. It should be noted here that ER-W3-2003-0366-02 provides guidance for isolating some of the permanent plant instrumentation that has the potential to be affected by SGCC. This instrumentation is all on the secondary system. The isolated instrumentation will therefore not be impacted by the SGCC. For the permanent plant instrumentation that has the potential to be impacted but that is not isolated during SGCC, the ER also justifies that there will be no harm done to this instrumentation. See the ER for further information on this subject.

Vibration analyses have also demonstrated that there will be no deleterious effect on the steam generators from the air sparging during the copper/lead step (Reference Dominion Engineering calculation C-4133-00-2).

Based on the above discussion, the probability of a malfunction of a component important to safety is not increased.

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

BASIS:

The consequences of an accident previously evaluated in the UFSAR will not be increased. As described above the SGCC will be done in Mode 5 prior to the start of RF 12. During this time the RCS pressures and temperatures will be well below those assumed in the UFSAR Chapter 15 accident analyses. The relatively low energy of the plant systems, when compared to full power operations, ensures that the consequences of any event that may occur are well bounded by the existing analyses. The consequences of these previously evaluated accidents would thus be much less severe if they were to occur during the steam generator chemical cleaning. Also, the assumptions used for the loss of shutdown cooling events bound the plant conditions that will be established for chemical cleaning. In addition, this process will not introduce or significantly increase any source terms that are associated with the UFSAR analyses. The stable plant conditions that are required during the chemical cleaning process also reduce the risk of any credible accident. Therefore, the consequences of any analyzed accident are not increased.

As stated in the answer to Question No. 1 above the break of a main steam line is evaluated in the UFSAR. The consequences of the UFSAR evaluated break were based upon the steam lines carrying only steam and specific chemicals normally added to the secondary system. Also as explained above, during the SGCC process the main steam line will also carry some small amounts of the chemical cleaning chemicals, ammonia and hydrazine. These

chemicals could have the potential to increase the consequences of the previously evaluated accident if the new chemicals could cause unanalyzed damage to safety related components that they may come in contact with during the postulated pipe break. Breaks in the FANP piping/hoses used for the SGCC also have the potential to introduce the chemical cleaning chemicals into the plant. This issue was discussed in the answer to Question No. 2 above. The conclusion is that there will be no impact on plant safety related components.

In addition, as discussed in the answer to Question No. 1 above, the chemicals used in the SGCC process are not very reactive at low temperatures. Therefore even if a leak were to occur the chemicals would soon cool to the ambient temperature (temperatures inside containment are expected to be higher than temperatures in the RAB but both will be sufficiently low to lower the chemical reactivity) and this would limit any impact on plant systems, structures or components. Systems, structures or components that could potentially be exposed to the chemicals, due to a piping rupture, will therefore suffer no consequences that are not bounded by the existing analyses in the UFSAR.

Also as discussed in the answer to Question No. 2 above Auxiliary Component Cooling Water (ACCW) Pump A is immediately adjacent to the inside of the wall penetration (elevation -27) through which the 4" FANP process line enters the Reactor Building. If the process line were to break and spray chemicals directly onto the pump motor, and if the chemicals were to enter the motor and contact the copper components in the motor, the motor would be severely damaged or destroyed. To protect the motor a scaffold barrier will be erected over the ACCW pump to prevent chemicals from entering the motor.

Upon return to full power operation, the consequence of an accident previously evaluated in the UFSAR is not increased since all changes to the secondary system, as a result of the SGCC process, will be within the design allowable limits. Any corrosion that will occur to the steam generators or affected plant systems will be well within design corrosion allowances. As stated above these effects are addressed in a separate 10 CFR 50.59 Report. The integrity of the affected plant systems will thus be maintained. Also, the removal of deposit from the steam generator tubes will help to ensure steam generator integrity by cleaning possible crack initiation and propagation sites. It should be noted however that the post SGCC eddy current test results will probably contain anomalies with reference to the previous eddy current results. This is because the SGCC will remove deposits from the tube surfaces that may currently be masking existing cracks in the tubes. Following the removal of the deposits these existing defects may become more readily detectable. This masking effect was first seen at the Millstone 2 plant in 1985. Nuclear Regulatory Commission Information Notice 85-37 contains information on this phenomenon. At Millstone essentially all of the masked defects were found within the region of the sludge pile (within thirteen inches of the tube sheet). This masking phenomenon should not be as extensive at Waterford 3 due to the smaller amounts of sludge in the

Waterford 3 steam generators relative to the amount at Millstone. Nevertheless it is expected that Waterford 3 could experience the masking phenomena during eddy current testing following the SGCC. As stated above steam generator eddy current testing will be performed in accordance with Technical Specification 3.4.4.

The chemical cleaning evolution has been developed to ensure effective chemical cleaning of the steam generators while minimizing corrosion attack. Laboratory qualification testing (as addressed in a separate 10 CFR 50.59 Report) has demonstrated that corrosion of steam generator internal materials due to the chemical solvent is minimal. FANP document No. 51-5008453-00, "Waterford Unit 3 Material and Geometry Review shows that the corrosion of the plant piping (main steam, blowdown), valves (ADV, MSIV) and components is also minimal. Corrosion monitoring measurements that will be taken following the SGCC will ensure that all affected materials remain within their corrosion allowance. Eddy current testing of the steam generator tubing and the inspection of key steam generator components such as the egg crates will ensure equipment and component integrity following SGCC. Therefore, the consequence of any analyzed accident is not increased.

As described above in the answer to Question No. 1 manual isolation of the control room ventilation system assures the operability of the system and the protection of the Operators from adverse effects due to potential chemical fumes. Therefore, the control room environment is not affected by the implementation of this process and the consequences of a toxic chemical accident are not increased.

The proposed Special Test Instruction also includes instructions for ensuring that the quantity of emergency breathing air for the Control Room is adequate, prior to start of the SGCC. It also includes instructions that the breathing air compressors are not to be run during SGCC. This will ensure that the emergency breathing air supply is available and is not contaminated during SGCC.

During SGCC the Control Room Ventilation system will be placed in the "Isolate" mode. This will ensure the operability of the system per Technical Specification 3/4.7.6.2 by isolating the charcoal adsorber in the system from contamination and degradation due to contact with chemicals released during the SGCC. Note that there will be times during the SGCC when no chemical fumes will be released. During these times operation of the system may be resumed at the discretion of the Operations Department. This will enable the carbon dioxide levels in the control room to be maintained at acceptable levels.

Shield Building Ventilation (SBV) is to remain secured in Mode 5 when the Shield Building Hatch is closed. When in Mode 5 with the hatch open the Process Lead must verify that ammonia and hydrazine vapors are less than one part per million (PPM) at the hatch opening before starting SBV. If the vapors are equal to or exceed 1 PPM then SBV start will be delayed until System Engineering recommends a course of action. If SBV is in operation and vapors are equal to or exceed 1 PPM then SBV will be secured and

System Engineering will be notified to evaluate the need to sample the filter charcoal. This will ensure compliance with Technical Specification 3/4.6.6.1 (Note: this TS is only applicable in Modes 1, 2, 3 and 4 and SGCC will be performed in Mode 5). If a radiological release occurs the system will be allowed to run normally, whether or not there are chemical fumes present. This is shown to be acceptable per the following discussion:

The Loss of Coolant Accident (LOCA) and Non-LOCA scenarios must be considered with respect to charcoal filter degradation affects on event consequences. For a lower mode LOCA (Mode 4 or Mode 5) Combustion Engineering provided an Infobulletin on Emergency Core Cooling System (ECCS) requirements needed to mitigate the consequences of a LOCA (ABBCE Infobulletin No. 99-01). This Infobulletin stated that with one High Pressure Safety Injection (HPSI) train available within 10 minutes, a severance of the largest line connected to the RCS will not uncover the core or increase the fuel cladding temperature. This means that the typical design basis LOCA radiological source terms will not be present and the safety charcoal filters will not be needed to meet the typical LOCA design basis requirements. Thus the issue of potential charcoal filter degradation is not predicted to produce adverse consequences for the Mode 4 and 5 LOCA due to no predicted fuel failure.

For the lower mode Non-LOCA events, the expected RCS and SG activities (RCS < 0.01 uCi/gm as of 9/27/00) are a factor of 100 below those used in the accident analyses. Non-LOCA events initiated from lower mode operation are not predicted to incur fuel failure. Thus it is expected that potential charcoal filter degradation would have no affect on the accident consequences.

During SGCC the Controlled Ventilation Area System Filter Trains will be secured. Technical Specification 3/4.7.7 is only applicable in Modes 1, 2, 3 and 4. If the CVAS Filter Train must be placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV system. Also as described above the system will be placed in service in the event of a radiological release regardless of chemical vapors. Chemical monitoring will be conducted for chemicals at the outside air intake for the RAB Normal Supply System and the Reactor Auxiliary Building (RAB) – 4 wing area. Sampling of the charcoal in the Controlled Ventilation Area System (CVAS) will also be conducted after the completion of the SGCC if monitoring results indicate that sampling is required. Also as stated above chemical fume degradation of the charcoal would have no affect on the accident consequences for a Mode 4 or Mode 5 accident (SGCC is performed in Mode 5).

During SGCC the Fuel Handling Building (FHB) Emergency Ventilation and Normal Ventilation will be secured. If the FHB Emergency Ventilation must be placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV and CVAS systems.

For all of the above-mentioned safety related, charcoal containing, HVAC

systems the monitoring during SGCC in conjunction with potential charcoal sampling (if high enough levels of chemical fumes are detected) will ensure that there is no degradation during the proposed SGCC. As stated above and in the answer to Question No. 1 all applicable safety related charcoal adsorber Technical Specifications will also be maintained during and after the SGCC. The safety-related charcoal adsorber beds will therefore be capable of performing their safety function during and following the proposed SGCC. Hence the consequences of a radiological release accident will not be increased due to charcoal degradation due to the proposed SGCC.

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 FSAR? ☐ Yes ☒ No

BASIS:

The consequences of a malfunction of equipment important to safety will not be increased. Cleaning will take place during Mode 5. Both shutdown cooling trains will be operable and available for heat removal. Both steam generators will be completely drained in Mode 5 and hence will not be operable but the two shutdown cooling trains will be operable and available for heat removal. Since the steam generator chemical cleaning process will have minimal impact on the fission product barrier of the steam generator tubes and tubesheet (see separate 10 CFR 50.59 "Process" Exemption), the potential failure of this barrier and the radiation release consequences associated with that failure will not be changed by the proposed SGCC. As discussed above the plant conditions (pressure, temperature, core decay heat) during the steam generator chemical cleaning process will be much less than the conditions assumed in the UFSAR Chapter 15 accident analyses. Any release would thus be bounded by the previous analyses, ie the UFSAR analyses that evaluate loss of SDC events, loss of off-site power, station blackout, toxic gas release, steam generator tube rupture and failure of the steam generator secondary side, bound the consequences of any credible accident scenario associated with chemical cleaning. Laboratory testing has demonstrated that there will be no adverse impact on the steam generator tubes and that the amount of material removed from the other internal steam generator components will be within the design allowances (see separate 10 CFR 50.59 "Process" Exemption). A separate 10 CFR 50.59 "Environmental" Exemption will evaluate the SGCC impact on the potential radiation release due to discharging the spent chemicals (including the potentially radioactive dissolved steam generator deposits) to the FANP waste tanks.

All chemical cleaning equipment will be removed from the site and all involved plant systems will be returned to their original condition except that the steam generators will have less corrosion products and will have been exposed to the chemical cleaning solutions. Exposure to chemical cleaning solutions does result in minimal corrosion of the steam generators; however, corrosion will be monitored, and the corrosion increase due to chemical cleaning will not be allowed to exceed the design corrosion allowances. As stated above this is addressed in a separate 10 CFR 50.59 "Process" Exemption. Therefore, the probability of a malfunction or an increase in the consequences of a

malfunction of equipment important to safety is not increased.

The proposed procedure changes have no impact on the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR.

In Mode 5, the steam generators are not required as a heat sink if both Shutdown Cooling Trains are operable (per Technical Specification 3.4.1.4). The steam generators function as a heat sink for the RCS. However, while in Mode 5 during Steam Generator Chemical Cleaning, the SDC system will act as the RCS heat sink. Since the steam generators may be drained during portions of the SGCC, increased reliance is placed on the SDC system. The proposed change does not alter the SDC system or the operation of the SDC system. The proposed change does not alter the redundancy and reliability of the SDC system. Since SDC reliability is unaffected by the proposed change, the consequences of a malfunction of the steam generators are not affected by the proposed change. Additionally, as discussed in the answer to question No. 1 above, in the event that shutdown cooling is lost when the steam generators are inoperable the SGCC will be terminated and auxiliary feedwater will be used to restore steam generator operability. Also as described above steam generator structural integrity is unaffected by the proposed SGCC. Therefore the radiological consequences of accidents evaluated in the SAR are unaffected.

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

BASIS:

The possibility of an accident of a different type than any previously analyzed in the UFSAR is not created. The UFSAR Chapter 15 analysis bound any potential accident associated with chemical cleaning. Spills of the chemical are bounded by toxic release analysis (addressed in a separate 10 CFR 50.59 "Environmental" Exemption). The chemicals associated with chemical cleaning are organic in nature and are readily absorbed by charcoal filtration. In addition, the chemicals released by the SGCC are the same as the chemicals that can potentially be released offsite by the nearby chemical plants as identified in the UFSAR. There are thus no "new" chemicals associated with SGCC. As stated above (see the answer to Question No. 1) monitoring and sampling of the charcoal, if chemicals are detected by the monitoring, will ensure the operability of the safety related HVAC systems. Also as stated in the answer to Question No. 1 above the impact of the SGCC chemicals on the blowdown lines will be acceptable.

Radiological release of an outside tank's contents is bounded by the radiological analyses performed in Chapter 11, 12 and 15 of the UFSAR (addressed in a separate 10 CFR 50.59 "Environmental" Exemption). The UFSAR chapter 15 analyses bound all other credible events associated with chemical cleaning of the steam generators, including toxic gas release.

The qualification testing and systems evaluation that was performed during

the development of this process ensured that no new accidents or malfunctions that were outside of the UFSAR analyses were generated. In fact, the chemical cleaning process equipment can be quickly and easily isolated from the permanent plant equipment if necessary. This feature allows immediate mitigation of any event that may be initiated by the temporary chemical cleaning equipment. In addition, laboratory testing has shown that it is also acceptable for the chemical cleaning solution to remain in a steam generator for approximately twenty-four hours if necessary. The chemical reaction is self limiting in that only the amount of chemicals necessary to remove the maximum expected deposits are added to the steam generators. Once the chemicals react with the deposits they are used up and any further reaction is minimal. As discussed previously, this is further addressed in a separate 10 CFR 50.59 "Process" Exemption. Therefore, the possibility of an accident of a different type than any previously evaluated in the UFSAR is not created.

An accident of a different type than those evaluated in the UFSAR will not be created. The only change in the plant is the way the Unit is operated during the Mode 5 SGCC sequence. All previously postulated Chapter 15 events, whether they are increase in heat removal events, or decrease in heat removal events, etc., bound the events which could possibly occur during the SGCC. Most of the events presented in the UFSAR correspond to events that can occur with the Unit at 100% power. At lower modes, such as Mode 5, only steam line failure events could have significant consequences and as discussed above this type of failure is currently evaluated in the UFSAR.

As described above in Description of Proposed Change the proposed Waterford 3 Steam Generator Chemical Cleaning Project requires numerous new system connections between the temporary FANP supplied equipment and permanent Waterford 3 components. Also as described above the new connections have been authorized by previously approved Procedure PMC-004-008, "Installation Procedure Steam Generator Chemical Cleaning Equipment Installation". The new connections that have not been previously approved are the Blowdown Connections.

As stated above the blowdown connections are in the non safety portion of the blowdown system and can be easily isolated from the safety related portions of the blowdown system. These connections will be used to inject the chemicals into the blowdown piping (through piping/hoses running into the plant from the FANP tank farm) and then into the steam generator through the blowdown ring. These connections therefore have no impact on the safety related portions of the system.

ER-W3-2003-0366-02 provides an evaluation of the impact on the permanent plant instruments that are not isolated (are exposed to the cleaning chemicals) during SGCC. It also evaluates the impact of isolating various permanent plant instruments during SGCC. One of the instruments that is isolated and evaluated in the ER is the SG blowdown radiation monitor. See the ER for additional information.

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 FSAR? ☐ Yes ☒ No

BASIS

There are no other potential malfunctions based on SGCC process implementation. Equipment reviews, calculations, laboratory testing and justifications have been performed. The conclusion is that there is no compromise of the pressure boundary piping, valves, seals, or vessels installed as permanent plant equipment either during or after SGCC. ER-W3-2003-0522-00 evaluates the consequences of introducing SGCC chemicals into the primary system due to steam generator tube leakage between the secondary and primary systems (at times during the SGCC the pressure in the secondary system will be higher than the pressure in the primary system and there is existing steam generator tube leakage in SG No. 1). The conclusion of the ER is that the expected chemical concentrations in the primary system are too low to cause any problems with the RCS materials or fuel. All credible plant equipment malfunctions have been addressed throughout the SGCC 10 CFR 50.59 evaluations/reports (as discussed above the three applicable evaluations/reports are the "Implementation" Evaluation, the "Process" Exemption and the "Environmental" Exemption).

Failure modes of the operation of the temporary chemical cleaning equipment is addressed in a separate 10 CFR 50.59 Report (see the 10CFR50.59 Environmental Exemption).

As discussed in the answer to Question No. 1 above ER-W3-2003-0366-02 provides guidance for isolating some of the permanent plant instrumentation that has the potential to be affected by SGCC. The isolated instrumentation will therefore not be impacted by the SGCC. For the permanent plant instrumentation that has the potential to be impacted but that is not isolated during SGCC, the ER also justifies that there will be no harm done to this instrumentation. The ER also evaluates the effect of the steam generator vacuum drying on the permanent plant instrumentation.

The steam generator blowdown piping from the steam generators to the containment isolation valves is Nuclear Safety Class 2. Outside of the containment the blowdown piping is Non Nuclear Safety. As stated in the answer to Question No. 1 above a break in the steam generator blowdown piping in not evaluated in Chapter 15 of the UFSAR. The potential break of the blowdown piping is bounded by the break of a main steam line. The main steam line break is bounding because of the relative size differences (forty-two inches for the main steam line vs. two inches for the blowdown line) of the two lines. A potential break in the safety class 2 blowdown piping caused by SGCC would therefore be a potential malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR. As discussed in the answer to Question No. 1 above the following FANP documents show that the impact, from SGCC, on the blowdown piping will be acceptable:

1. "Waterford Unit 3 Material and Geometry Review and Evaluation" (FANP Document Identifier 5008453-00).
2. "PVNGS – 2 Chemical Cleaning Materials Evaluation / Corrosion Estimates" (FANP Document Identifier 51-1228499-00, referenced in FANP Document 5008453-00).
3. "PV-1 HTCC Material Evaluation" (FANP Document Identifier 51-1234952-00, referenced in FANP Document 5008453-00).
4. "Salem Unit 2 Materials Evaluation" (FANP Document Identifier 51-1269270-00, referenced in FANP Document 5008453-00).
5. "300 F Chemical Cleaning Velocity Test Report" (FANP Document Identifier 51-1234107-00, referenced in FANP Document 51-1234952-00).
6. "PVNGS-1 High Temperature Chemical Cleaning Research Qualification Test" (FANP Document Identifier 51-1234937-00, referenced in FANP Document 51-1269270-01).
7. "PV-1 High Temperature Chemical Cleaning Optimization Test Report" (FANP Document Identifier 51-1234949-00, referenced in FANP Document 51-1269270-01).
8. Dominion Engineering, Inc. Report R-4135-00-1, Rev. 0, "Summary of Nominal Process and Low Deposit Loading Autoclave Tests for Qualification of the Siemens Chemical Cleaning Process at Waterford 3", August 2000
9. "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report", (FANP Document Identifier 51-5030944-00), September 29, 2003.
10. Dominion Engineering,Inc. Letter L-4160-01-2, "Material and Geometry Review – Waterford 3 Steam Generator Chemical Cleaning", September 15,2003.

A break in the safety related blowdown piping (or the non safety related blowdown piping) can therefore not be created by the proposed Steam Generator Chemical Cleaning.

Also as discussed in the answer to Question No. 1, 2 and 3 above the effect of the cleaning chemicals (either liquids or vapors) has been evaluated and determined to cause no harm to any equipment, both inside and outside of containment, that is important to safety. There is therefore no effect, even from the UFSAR postulated mainsteam line break, on plant safety related structures, systems and components.

Vibration analyses have also demonstrated that there will be no deleterious effect on the steam generators from the nitrogen/air sparging during the SGCC (Reference Dominion Engineering calculation C-4133-00-2).

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

BASIS:

The steam generator tubes are a fission product barrier as described in the FSAR. The steam generator tubes are part of the reactor coolant system pressure boundary. The fuel cladding and the main steam isolation valves

(containment isolation valves) are also fission product barriers described in the FSAR.

Steam Generator Chemical Cleaning (SGCC) is an established foreign and domestic nuclear industry practice. The Electric Power Research Industry endorses chemical cleaning (ref EPRI TR-104553) and a body of Steam Generator chemical cleaning experience exists. The Waterford 3 SGCC process was laboratory tested in the FANP laboratory and confirmatory testing was performed in a separate laboratory by Dominion Engineering (see References above in the answer to Question No. 6). As discussed above the proposed SG cleaning chemicals will not adversely impact the S/G Inconel 600 tube material (see separate 10CFR50.59 Exemption Report), i.e. there will be no adverse corrosion of the tubes. Therefore the proposed chemical cleaning will not significantly decrease the wall thickness of the steam generator tubes and there will be no adverse impact on this fission product barrier.

Qualification testing and material evaluation studies (as described herein and in the "Process" 10 CFR 50.59 Exemption) have demonstrated that the base metal corrosion of the steam generator components and plant systems (including main steam isolation valves) that are exposed to chemical solvent, in both liquid and vapor form, is minimal and well within the design corrosion allowances. The proposed SGCC therefore has no impact on the steam generator components and plant systems that are exposed to the cleaning chemicals and decomposition products.

Following completion of the SGCC the steam generator tubes will be eddy current tested to ensure steam generator integrity. Any tube that has been determined to be defective will be removed from service. As stated in the answer to Question No. 2 above there has been a previous history, in the nuclear industry, of steam generator deposits masking defects in the steam generator tubes from detection during eddy current testing. It is expected that this same phenomena will occur at Waterford 3 during the eddy current testing that will be conducted following the proposed SGCC. These potential defects however, will not be caused by the Waterford 3 SGCC. The potential defects will have been pre existing and will merely be made detectable by the steam generator chemical cleaning process (steam generator tube deposit removal). The pre SGCC laboratory qualification testing, corrosion monitoring and post SGCC tube eddy current testing all serve to ensure compliance with Technical Specification 3/4.4.4. See the separate 10 CFR 50.59 "Process" Exemption for further information on the effects of the proposed SGCC on the steam generator internals. Therefore, structural integrity of the steam generator tubes is assured and the design basis limits for the steam generator tubes as defined in Technical Specification 3/4.4.4 are not reduced.

No temporary Steam Generator Chemical Cleaning equipment will be used inside containment.

The fuel cladding is also a fission product barrier as described in the FSAR. ER-W3-2003-0522-00 addresses the impact of SGCC chemicals on Fuel

and Reactor Coolant System Components due to leakage, through the steam generator tube(s), from the secondary to primary side during the chemical cleaning. The conclusion of the ER is that there will be no impact on either the fuel of reactor coolant system components.

Based on the information provided in this evaluation and the References, the chemical cleaning of the Waterford 3 steam generators does not pose any significant impact on plant nuclear safety, is well bounded by the safety analyses of the UFSAR and does not reduce the safety as defined in the basis of the Technical Specifications. The chemical cleaning process has been qualified and designed for implementation such that it will not result in safety issues and will have an overall minimal impact on the plant. The process will remove deposits from the steam generator tubes in such a way as to not degrade the structural integrity of the components or impact the health and safety of the public. Therefore the design basis limits for the fuel cladding, RCS pressure boundary and containment are not reduced by the proposed chemical cleaning.

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

BASIS:

The proposed Refuel 12 Steam Generator Chemical Cleaning is a maintenance activity to remove iron and copper/lead deposits in the Waterford 3 steam generators. If not removed the copper/lead deposits have the potential to interfere with the eddy current inspection of the steam generator tubes as required by Technical Specification 3/4.4.4. The proposed SGCC does not impact the design basis of the steam generators. No steam generator design basis calculations are impacted by the proposed change.

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 Steam Electric Station

Document Reviewed: ER-W3-2003-0585-000 **Change/Rev.:** N/A

System Designator(s)/Description: N/A

Description of Proposed Change

1. Revise certain subsections of UFSAR Section 15.4.3.1, "Inadvertent Loading of a Fuel Assembly into the Improper Position" and related tables and figures to make them consistent with the latest fuel misloading analysis and update information about details of the fuel manufacturing process, fuel movement, and core verification .
2. Revise TRM Section 3.3.3.2 and the Bases for this section to incorporate new requirements for incore instrumentation operability.
3. Revise the INCA software (Program NSFICDINCA.FOR) to address the new incore instrumentation operability requirements.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: 03-013-1)	Sections I, II, III, IV, and VI required

Preparer: *Paul M. Melancon/* Paul M. Mel /*EOI-W3/Nuclear Engineering/* 11-15-03
Name (print) / Signature / Company / Department / Date

Reviewer: *Edmond G. Wiegert/* Edmond G. Wiegert /*EOI-W3/Nuclear Engineering/* 11-15-03
Name (print) / Signature / Company / Department / Date

OSRC K. WALSH K. Walsh 11/16/03
Chairman's Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.9) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
C.R. Bergeron (NCIS)	Input on revision to the INCA software

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☒	☐	*TOC pages 15-x, 15-xi, 15-xxviii, and 15-xxix; 7.7.1.7; 15.4.3.1.1.1; 15.4.3.1.1.2; 15.4.3.1.2.1; Delete Tables 15.4-18, 18a, and 18b; Add Tables 15.4-37, 38; Delete Figures 15.4-62, 63, 64, 65, 65a, and 65b, Add Figures 15.4-98, 99*
TS Bases	☐	☒	
Technical Requirements Manual	☒	☐	*Section 3.3.3.2, Section 3/4.3.3.2 Bases*
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V <u>OR</u> perform a 50.59 Evaluation per Section VI <u>AND</u> initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculation Manual[3]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V <u>OR</u> perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. **Basis**

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

Condition Report CR-WF3-2002-00253 identified that the existing fuel misloading analysis (performed for licensing Waterford) includes assumptions that are not assured through plant practices. Specifically, CEA symmetry testing is not performed during post-refueling startup testing and the Incore Nuclear Instrumentation (INI) System does not always have 100% of its detectors operable. It has been confirmed that the fuel misloading analysis performed for the extended power uprate project (Westinghouse Calculation CN-WFE-CWTR-03-4) is applicable to Cycle 13 as documented in Westinghouse Letter NF-03-W-WTFD-47 dated November 13, 2003. This analysis, which will be credited for Cycle 13, does not require CEA symmetry testing but assumes more restrictive INI System operability requirements than are presently in the Technical Requirements Manual (TRM). Therefore, the TRM will be revised to incorporate the new requirements. In addition, changes to certain subsections, tables, and figures associated with UFSAR Section 15.4.3.1, "Inadvertent Loading of a Fuel Assembly into the Improper Position", are necessary to remove statements about activities that are no longer performed and to correct/clarify information so that the UFSAR is consistent with current plant practices. Also, FSAR Section 7.7.1.7, which discusses the INI System, will be revised appropriately.

The Waterford 3 licensing basis is not affected by this activity and no Technical Specification changes are required. No other licensing basis documents are impacted. No tests or experiments are involved with this change.

C. **References**

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LBDS_50_59 (Group)	*Misload (3 hits), misloading (2 hits), "inadvertent loading" (11 hits), "fuel misloading" (1 hit),"fuel misload" (0 hits), "loading errors" (1 hit), "loading error" (0 hits), "core misloading" (0 hits), "core misload" (0 hits), "improper core loading" (0 hits), "improper loading" (0 hits), "assembly misloading" (1 hits), "assembly misload" (0 hits), "assembly loading errors" (0 hits), "incore" (44 hits), ICI (5 hits), "rhodium" (5 hits) "INCA" (5 hits),*

LBDs/Documents reviewed manually:

UFSAR Section 7.7.1.7, UFSAR Section 15.4.3.1 and associated subsections, tables, and figures; Waterford 3 SER Section 15.2.4.5, SRP Section 15.4.7; Condition Report CR-WF3-2002-00253;Technical Requirements Manual, Westinghouse Calculation CN-WFE-CWTR-03-4, Westinghouse Letter NF-03-W-WTFD-47

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☐ Yes
☒ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ☐ No

Name of Security Plan reviewer (print) / Signature / Date

V. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A, above.

A. Check the applicable boxes below. If any of the boxes are checked, a 50.59 Evaluation is not required. If none of the boxes are checked, perform a 50.59 Evaluation in accordance with Section V. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.6.1.1:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; AND

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; AND

The proposed activity does not adversely affect a method of evaluation that demonstrates intended functions of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed change already exists per Section 5.6.1.2. Reference 50.59 Evaluation # _____ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.6.1.3. Reference: _____

☐ The proposed activity is controlled by another regulation per Section 5.6.1.4.

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10CFR50.59 Review Program Guidelines for guidance.

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

An additional requirement for incore instrumentation operability is being added to TRM Section 3.3.3.2 and the Bases for this section is being enhanced. The INCA software, used to assess operability of the incore instrumentation system, is also being revised. Certain subsections, tables, and figures of UFSAR Section 15.4.3.1, "Inadvertent Loading of a Fuel Assembly into the Improper Position" are being updated as well as FSAR Section 7.7.1.7 which presents information about the Incore Nuclear Instrumentation (INI) System.

Reason for proposed Change:

Since CEA symmetry checks are no longer performed at Waterford 3 and 100% incore detector operability is not maintained (identified in CR-WF3-2002-00253), a revised fuel misloading analysis for Cycle 13 was credited. The additional requirement added in the TRM for operability of the INI System is necessary to satisfy the requirements of the new fuel misloading analysis. The revision to the INCA software is required to facilitate monitoring of the new incore instrumentation operability requirement by Operations. The FSAR changes are necessary to provide updated information based on the revised fuel misloading analysis and to update and enhance information about details of the fuel manufacturing process, fuel movement, and core verification after core loading.

50.59 Evaluation summary and conclusions

This evaluation determined that the changes to the TRM and to the INCA software for incore operability are acceptable and ensure that the assumptions of the fuel misloading analysis for Cycle 13 remain valid. It was determined that accident consequences are not increased and that no methodology changes have been identified related to the new fuel misloading analysis. The proposed UFSAR changes are appropriate and consistent with the Waterford 3 licensing basis.

B. License Amendment Determination

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 FSAR? ☐ Yes ☒ No

BASIS:

No changes to plant systems or components are being made. The proposed UFSAR changes update information about the fuel manufacturing process, clarify information about fuel movement practices at Waterford 3, and revise the information presented about the fuel misloading event based on the new fuel misloading analysis for Cycle 13. The new analysis does not impose plant operational requirements that would change the frequency of occurrence of a misload event. Information in the FSAR related to CEA symmetry testing (which is no longer performed at Waterford 3) is being deleted since the testing is not a requirement of the new misloading analysis. The performance of CEA symmetry testing was intended to be used for detection of an incorrect core loading configuration, however, this testing does not affect whether a fuel misloading event occurs. The UFSAR changes related to the fuel manufacturing process only clarify identification and record-keeping details and do not reduce the capability of the manufacturing process to produce fuel assemblies in accordance with design requirements. As stated in UFSAR Section 15.4.3.1.1.1, extensive quality control and quality surveillance programs are in place to ensure that fuel is built correctly. The UFSAR changes related to fuel movement practices at the Waterford 3 site enhance and clarify the description provided in the UFSAR but do not alter the potential for mispositioning a fuel assembly during fuel movement activities. Fuel movement procedures and core verification requirements are not changed. During refueling outages at Waterford 3, fuel movement is conducted in strict compliance with station procedures using written as well as visual indicators to track the location of fuel and CEAs. As a barrier to help prevent the operation of a core that has been misloaded by incorrect placement or orientation of one or more fuel assemblies, a "full core verification" is performed after fuel movement in the core is complete during a refueling outage. This effort involves the visual identification of each fuel assembly in the core by its serial number to confirm that the loaded configuration is in accordance with applicable reload design documentation. Any loading errors found are corrected prior to reactor reassembly. Performance of the core verification is a procedural requirement and the work is independently verified. The core verification process is usually videotaped and reviewed by Reactor Engineering. In addition, during core verification, rodded core locations are confirmed to each contain a Control Element Assembly. This process provides multiple barriers to prevent a fuel assembly misloading.

The additional requirement being added to the TRM and the associated INCA software change do not alter the Incore Nuclear Instrumentation (INI) System or cause it to operate differently. These changes only impose an additional requirement for system operability and provide for monitoring that requirement.

Based on the above, the frequency of occurrence of a fuel misloading event at Waterford 3 is not increased.

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 FSAR? ☐ Yes ☒ No

BASIS:

No changes to plant equipment result from the activities being evaluated herein. The revised fuel misloading analysis to be credited for Cycle 13 does not include any assumptions or specify any requirements that would change the function or operation of plant equipment or systems. The incore instrumentation system operability requirement added to the TRM and the INCA software change only impose an additional requirement for system operability and provide for monitoring that requirement. The INCA software is an on-demand software module of the PMC which just reads data from the PMC, performs the incore instrumentation operability assessment, and prints out the results for plant operators. Neither the current INCA software nor the revised version performs any control functions, drives any annunciators, or issues any alarms. No structures, systems, or components important to safety are affected. Neither the incore instrumentation system nor any fuel handling equipment is being modified. In addition, procedures for operating these systems are not being altered. Therefore, there is no increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety evaluated in the FSAR.

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

BASIS:

The only accident presented in Chapter 15 of the FSAR that is potentially affected by no longer performing CEA symmetry testing and operating with less than 100% of the incore instrumentation is the fuel misloading accident (FSAR Section 15.4.3.1). The proposed UFSAR changes related to the fuel manufacturing process and fuel movement practices at Waterford 3 provide clarification only and do not impact the consequences of a potential fuel misloading event or any other accident described in FSAR Chapter 15. The deletion of information related to CEA symmetry testing is necessary since this testing is no longer performed at Waterford 3. The revised fuel misloading analysis being credited for Cycle 13 does not require CEA symmetry testing for misload detection but assumes more restrictive INI System operability requirements than are presently in the Technical Requirements Manual (TRM). Therefore, the TRM will be revised to incorporate the new requirements so that misload detection capability is adequate to satisfy the assumptions of the analysis. Specifically, it will be necessary to ensure that at least one incore instrument string in each 4 x 4 array of fuel assemblies in the core has at least three functional rhodium detectors (one each at any three of the five axial levels). In addition to the TRM change, the INCA software on the Plant Monitoring Computer (PMC) is being changed to update its assessment logic and reporting capability for operability of the INI System so that Operations can ensure that the new TRM requirements are met. During startups after refueling outages, procedure NE-002-110, Fuel Symmetry Verification, is performed prior to exceeding 30% power to verify that no detectable fuel misloadings exist. This procedure requires that the INI System be operable in accordance with TRM Section 3.3.3.2.

Operation with the revised INI operability requirements will ensure that the assumptions of the fuel misloading analysis credited for Cycle 13 remain valid. The analysis explicitly considered the power distribution and thermal margin resulting from several potential worst case misloads and concluded that the Specified Acceptable Fuel Design Limits (SAFDLs) were not exceeded by the worst undetectable misload. Hence, no fuel failure is expected to occur as a result of a fuel misloading event and the dose consequences of a fuel misloading event are not changed from those reported 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 FSAR? ☐ Yes ☒ No

BASIS:

As stated in the basis for the answer to Question 2 above, no changes to any equipment are being proposed by this activity. The revised fuel misloading analysis to be credited for Cycle 13 does not include any assumptions or specify any requirements that would change the function or operation of plant equipment or systems. The analysis shows that fuel failure is not expected as a result of the representative worst case misloading event. The INCA software does not provide input to other software or hardware and it does not perform any control functions now or after adding assessment capability for the new incore instrumentation operability requirements specified in the revised TRM. Therefore, since fuel integrity is maintained and no equipment is being changed and no interfaces are being created that would cause any equipment or systems to operate in a different manner than previously described in the FSAR, there is no increase in the consequences of a malfunction of SSCs important to safety.

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

BASIS:

As stated previously, no changes are being made to equipment or procedures. The revised INCA software will be used in the same way as before it was revised. The TRM change only imposes an additional requirement on INI System operability. Thus, the possibility of a new fuel misloading event not previously evaluated will not be created. Since the proposed changes to the FSAR, TRM, and INCA software do not affect the design or operation of any equipment or systems, this activity will not initiate an accident. Therefore, no accident of a different type than has previously been evaluated in the FSAR will 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 FSAR? ☐ Yes ☒ No

BASIS

As stated in the basis for the answer to Question 2 above, no changes to any equipment or systems are being proposed. The changes to the FSAR and the TRM are document changes only and do not involve any interaction with systems or components. The function of the revised INCA software will be the same as the current software in that it will provide an incore instrumentation operability assessment to plant operators but perform no control or alerting functions. Hence, no new malfunctions of SSCs are created by this activity and, therefore, no malfunctions with different results than previously evaluated in the FSAR are introduced.

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

BASIS:

The fission product barrier of concern in this activity is the fuel cladding. The fuel misloading analysis for Cycle 13 considered several potential misloading configurations. The representative worst case misload was identified and it was determined that the SAFDLs would not be exceeded should such a misload occur. The TRM and associated INCA changes ensure that the assumptions of the analysis remain valid. Therefore, no cladding damage is expected as a result of this worst case misload.

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

BASIS:

The method used for analysis of the fuel misloading event at Waterford 3 was to analyze a worst-case fuel misloading configuration that is not detectable and to show that the SAFDLs would not be exceeded should such a misload occur. The method of evaluation for fuel misloading is not changed since a fuel symmetry test is still simulated in the revised analysis as was the case in the previous analysis and only the testing practices at the plant have changed (see Westinghouse Letter NF-03-W-WTFD-47). Note that the specific methodology for analysis is not described in the FSAR.

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 Steam Electric Station

Document Reviewed: **Change/Rev.:** N/A
Cycle 13 Core Operating Limits Report (COLR) [Westinghouse Letter NF-03-W-WTFD-42]; Core Operating Limits Supervisory System (COLSS) and Core Protection Calculator (CPC) System Addressable Constant changes for Cycle 13 [Westinghouse Letter NF-03-WTFD-40]; and UFSAR changes for Cycle 13 as described herein.
System Designator(s)/Description: N/A

Description of Proposed Change

This evaluation addresses the changes described below that support implementation of Cycle 13.

(Continued on next page)

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: _____) 03-014	Sections I, II, III, IV, and VI required

Preparer: P.M. Melancon/ P.M. Mel /EOI-W3/Nuclear Engineering/ 10-29-03
Name (print) / Signature / Company / Department / Date

Reviewer: D.E. Barr/ /EOI/NEAD Fuel Fabrication/ 10/29/03
Name (print) / Signature / Company / Department / Date

OSRC 11/3 4/03
Chairman's Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.9) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
D.E. Barr (Fuel Fabrication Coordinator, Waterford 3, See References 37 & 38)	Fuel Mechanical Design and Fabrication
S.L Rowe (NEAD, Echelon, See References 39 & 40)	Core Physics Design, Fuel Management

Description of Proposed Change (continued) (References cited are at the end of this evaluation)

Cycle 13 Non-Fuel Related Changes

The Waterford 3 Cycle 13 Safety Analysis Groundrules document (Reference 2) was provided to the fuel vendor (Reference 1) to be used as input for the Cycle 13 reload analyses. Changes to the Groundrules for Cycle 13 relative to Cycle 12 were primarily related to fuel management and core design. In addition, miscellaneous enhancements, clarifications, and reference updates were made to improve the Groundrules. The Groundrules were reviewed and approved in accordance with NOECP-702 (Reference 52). The following are the major non-fuel changes for Cycle 13:

Item	Change for Cycle 13	Reason for Change
1	Nominal Primary Coolant Flow changed from 107% to 110%.	Correction per CR 2003-00120
2	Maximum pressurizer spray flow rate changed from 375 GPM to 550 GPM.	Correction per CR 2003-0631
3	ADV Max Flow Rate (@949 psia) changed to ADV Max Flow Rate (@865 psia). Flowrate of 934,000 lbm/hr is unchanged.	Updated reference pressure based on startup data
4	ADV Min Flow Rate (@865 psia) changed to ADV Min Flow Rate (@900 psia). Flowrate of 583,200 lbm/hr is unchanged.	Updated reference pressure based on startup data
5	The power measurement uncertainty at 100% power when using the ultrasonic flowmeter installed during RF 11 (0.5%) was added to the Groundrules	Changed to take advantage of the more accurate device for analyses performed at 100% power

Cycle 13 Nuclear Design Changes

The Waterford 3 Cycle 13 core will contain 125 irradiated assemblies (Region T and Region U fuel), and 92 fresh Region W assemblies. This batch size is the same size as the reload batches for both Cycle 12 and Cycle 10. All 49 Region S assemblies and 43 of the 76 Region T assemblies present in the Cycle 12 core will be discharged from the core. The fuel pin initial U-235 enrichments in the fresh Region W fuel assemblies will range from 4.20 - 4.55 weight percent. These pin enrichments are very similar to the enrichments of 4.17 and 4.57 weight percent in the Region U fuel assemblies that were loaded in Cycle 12.

Integral burnable poison rods containing erbia were first introduced into Waterford 3 cores in Cycle 9 (Region R). Reload Region W will consist of six sub-batches with different erbium loadings. The loading of erbium will range from 24 erbia rods in sub–batch W1 to 100 rods at 2.1 weight percent erbium oxide in sub–batch W6. The total number of erbia rods (6176) in fresh fuel assemblies is the same as it was in the previous cycle. The elimination of the alumina spacer disks inside the fuel pins and the shortening of the lower end caps, changes which were introduced in Cycle 12 with Region U fuel, will also be employed in Region W. As a result, the Cycle 13 reload will result in moving the active fuel zone down by 0.355 inches in a larger fraction of the core than before.

The reactor core will be loaded with quarter-core rotational symmetry in a low neutron leakage configuration in which "twice-burned" Region T assemblies will be loaded primarily on the core periphery, and fresh Region W assemblies generally will be mixed with "once–burned" Region U assemblies in the core interior. This loading strategy is similar to that used for Cycle 12 when Region S assemblies were loaded on the edge of the core, and fresh Region U fuel assemblies were usually mixed with Region T

assemblies in the interior. This type of loading tends to minimize power peaking in the core and neutron fluence in the reactor vessel. The peripheral assembly integrated powers generally will be similar to those in Cycle 12 and the neutron flux at the excore detectors in Cycle 13 is predicted to be about 1.6 percent larger than in Cycle 12.

The Cycle 13 core was designed on the basis of a nominal, best estimate cycle energy of 514.5 EFPD for a Cycle 12 energy of 524 EFPD. Evaluations have been completed to demonstrate the applicability of the reload analyses for Cycle 12 energies between 514 and 544 EFPD, and for corresponding Cycle 13 energies from 545.1 EFPD down to 528.5 EFPD.

Predicted moderator temperature coefficients and critical boron concentrations are similar to those calculated for Cycle 12. The cycle maximum power peaking factors are expected to be very similar.

Cycle 13 Fuel Mechanical Design Changes

The Region W fuel design was evaluated against previous fuel designs and any plant changes in Cycle 13. This evaluation was performed using plant and Westinghouse supplied documentation and observation of the manufacturing process at Columbia.

The Inconel top grid, first used at Waterford 3 in Cycle 12 (installed on 32 of the 92 Region U assemblies) is installed on all 92 Region W assemblies. This top grid has been developed to improve the margin for top grid to rod fretting in the third cycle of operation. The assemblies with the Inconel top grid from Region U are not scheduled to be placed near the core periphery until Cycle 14. A small change has been made to the grid spring crown on the Inconel grids used for the Region W assemblies to prevent a rod scraping problem identified in the manufacture of a previous sub-batch.

There are no other major mechanical design changes for the Region W grid cages. A slight chamfer was added to the inboard end of the center and outer guide tubes to prevent a threading burr from forming during manufacture.

One minor documentation change was made associated with the fuel design. Specifically, the drawings that define the fuel and Erbia rods were split to separate the overall rod geometry from the pellet stack information.

Core Operating Limits Report (COLR) Changes

As a result of Cycle 13 safety analyses, the COLR is being revised for Cycle 13. The Cycle 13 COLR meets the requirements of Waterford 3 Technical Specification 6.9.1.11. The changes to the COLR for Cycle 13 will be consistent with the requirements of Reference 7, which support the analyses documented in the RAR (Reference 5). The changes for Cycle 13 are shown below:

Cycle 12 COLR Section or Figure	Cycle 13 Change
Section I Introduction	List changes to COLR for Cycle 13
Section 3.1.3.1.a and Section 3.1.3.1.b Movable Control Assemblies-CEA Position	Add note below these sections to explain that Section 3.1.3.1.a corresponds with Technical Specification Section 3.1.3.1.c and Section 3.1.3.1.b corresponds with Technical Specification Section 3.1.3.1.d
3.2.4 DNBR Margin	Revise notes after the text to clarify usage of Figures 8 and 9.

Cycle 12 COLR Section or Figure	Cycle 13 Change
COLR Figure 6, Allowable Peak Linear Heat Rate Versus Tc (COLSS In Service) COLR Figure 7, Allowable Peak Linear Heat Rate Versus Tc (COLSS Out Service)	Add vertical lines at each endpoint of the plotted line for clarity
COLR Figure 8, Allowable DNBR with Any CEAC Operable (COLSS Out of Service) COLR Figure 9, Allowable DNBR with No CEAC(s) Operable (COLSS Out of Service)	Revise DNBR limits to support increased margin requirements for the CEA Ejection Fuel Failure Analysis and the Single CEA Withdrawal (SCEAW) Within CPC Deadband Analysis Add Figure 8A to provide better resolution of the higher power ranges
Section III Methodologies	Add CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure", in accordance with License Amendment 191 (Reference 55)

COLSS/CPC Setpoint Changes

Certain events (such as Anticipated Operational Occurrences (AOOs) like CEA drops, single CEA deviations within the CPC deadband, and Excess Load with Loss of AC power) are analyzed to obtain the required overpower margin that needs to be set aside in COLSS/CPC to prevent fuel failure. Requirements resulting from the transient analyses are input into COLSS/CPC setpoints process and cycle-specific addressable constants are derived. The constants are modified prior to cycle startup to ensure that the provisions of the safety analyses are implemented for cycle operation. Reference 8 transmitted the required COLSS/CPC addressable constant changes for Cycle 13.

UFSAR Changes

Revisions to the Waterford 3 Updated FSAR (UFSAR) will be required as a result of Cycle 13 changes. Primarily, these changes include revisions to the description of the Cycle 13 core with the new assemblies, new core physics parameter values, and changes to the fuel design description. In addition, some editorial changes will be made. The following table lists the sections, tables, and figures to be changed.

Section/Table/Figure	*Description*
Page 4-i	Table of Contents
Page 4-iv	List of Tables
Page 4-ix	List of Figures
4.2.1.2.5	Fuel Rod Pressurization
4.2.2.1	Fuel Assembly
4.2.4.2.3.2	Rod Scanner
Table 4.2-1	Mechanical Design Parameters
4.3A	Fuel Cycle 12
4.3A.1	General Description
4.3A.2.1.1	Fuel Design
4.3A.2.3	Thermal Design
4.3A.2.5	Shoulder Gap Adequacy
4.3A.3.1.1	Fuel Management

Section/Table/Figure	Description
4.3A.3.1.2	Power Distribution
4.3A.3.1.3	Maximum Fuel Rod Burnup
Table 4.3A-1	Cycle 12 Core Loading
Table 4.3A-5	Cycle 12 Thermal Hydraulic Parameters at Full Power
Figure 4.3A-2	Waterford-3 Fuel Management Scheme
Figure 4.3A-3	Waterford-3 Cycle 12 Assembly Average Burnups
Figure 4.3A-7	Waterford 3 Part-Length CEA Insertion Limit vs Thermal Power *{Editorial change only}*
Figure 4.3A-8	Waterford-3 Cycle 12 Assembly Relative Power Density, BOC, HFP, Equilibrium Xenon, ARO
Figure 4.3A-9	Waterford-3 Cycle 12 Assembly Relative Power Density, MOC, HFP, Equilibrium Xenon, ARO
Figure 4.3A-10	Waterford-3 Cycle 12 Assembly Relative Power Density, EOC, HFP, Equilibrium Xenon, ARO
15.0.3.1.12	DNBR Propagation Analysis Method
15.1.2.4.2	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component

Summary of Major Reload Topics

The Cycle 13 Reload Analysis Report (RAR)[Reference 5] documents the analyses and assessments performed to demonstrate the acceptability of the Cycle 13 core design. This report was reviewed by Entergy personnel as required by NOECP-702, Waterford 3 Reload Process. Below are brief summaries of the purpose and results for the major areas of evaluation. Cited references are located at the end of this evaluation.

Physics Assessment

The purpose of the physics assessment is to confirm that the physics parameters used in the AORs are applicable to Cycle 13. The primary elements that impact the physics assessment are the Cycle 13 specific core design characteristics. These were explicitly incorporated into the neutronics models and files used to perform the physics assessment. The physics assessment was performed at a full power of 3441 MWth. It was determined that a cycle-specific analysis was required to be performed for the Fast Trip CEA Ejection event. In addition, it was found that setpoint adjustments would be required to provide adequate margin for the Single CEA Deviation Within Deadband event and the Excess Load with Loss of AC event.

Revised setpoints have been determined (Reference 8) and will be implemented per station procedures prior to the startup of Cycle 13. Also, the requirements of Table 6-9 of Reference 5 (PAC Assessment External Requirements) have been adequately addressed for Cycle 13 and are documented in Reference 53.

Fuel Performance Analysis

The purpose of the fuel performance analysis is to demonstrate acceptable fuel performance (e.g. rod internal pressure, power-to-melt, axial densification factor) for the Cycle 13 design and operating conditions. The reload process evaluates the applicability of the Fuel Performance Analysis of Record

(AOR) to Cycle 13. If determined to be applicable, the AOR results are valid for use by Mechanical Design, Non-LOCA Safety Analysis, ECCS Performance Analysis, and Digital Setpoints Analysis.

The AOR and related assessments for the Appendix K power uprate (Cycle 12) and the transition of manufacturing from Westinghouse Hematite to Westinghouse Columbia were determined to be applicable to Cycle 13. There is no change in the safety analysis or mechanical analysis related fuel performance results for Cycle 13.

Thermal Hydraulic Analysis

The purpose of the Thermal Hydraulic (TH) analysis was to demonstrate that thermal hydraulic performance of the Cycle 13 core remained bounded by the Thermal Hydraulic AORs. Steady state DNBR analyses for Cycle 13 at the rated core power of 3441 MWth were performed using the TORC computer code, the CE-1 Critical Heat Flux (CHF) correlation, simplified TORC modeling methods, and the CETOP-D code. The Modified Statistical Combination of Uncertainties (MSCU) methodology was applied with Waterford 3 specific data and other uncertainty factors at 95/95 probability/confidence levels to verify that the Specified Acceptable Fuel Design Limit (SAFDL) of 1.26 on the CE-1 Critical Heat Flux (CHF) correlation minimum DNBR remains applicable to Cycle 13. The effects of fuel rod bowing on DNBR margin are incorporated into the safety and setpoint analyses. NRC approved methods were used and it was determined that the TH performance of the Cycle 13 core is bounded by the AORs.

Non-LOCA Safety Analysis

The purpose of the non-LOCA safety analyses is to demonstrate that for the Cycle 13 design and operating conditions, the consequences of various postulated Design Basis Events (DBEs) are acceptable. The reload process evaluates the applicability of the AORs for the various DBEs to Cycle 13. The evaluation documents the key analysis inputs from the safety analysis groundrules, the bounding physics analysis, the bounding fuel performance analysis, and the bounding thermal-hydraulic analysis needed to validate the bounding non-LOCA safety analysis. The DBEs are categorized into three groups: Moderate Frequency, Infrequent, and Limiting Fault events. The DBEs were evaluated with respect to four criteria: offsite dose, reactor coolant system pressure, fuel performance [DNBR and fuel centerline melt Specified Acceptable Fuel Design Limits (SAFDLs)], and loss of shutdown margin. All Chapter 15 events were reviewed to assure that they meet their respective criteria for Cycle 13.

The results of the non-LOCA evaluations indicated that a cycle-specific assessment of the CEA Ejection fuel failure analysis based on DNBR criteria was required. This assessment was performed and it was demonstrated that the results presented in the FSAR remain bounding. However, it was necessary to credit higher DNB margins in the analysis and, therefore, COLSS setpoints and COLSS Out-of-Service Limit Lines in the COLR will be revised to preserve these margin requirements which assures that the Cycle 13 CEA Ejection fuel failure results are bounded by the DNB criteria limit of 9.12% fuel failure in the AOR. In addition to these setpoint change requirements, the non-LOCA assessment determined that a DNB (COLSS or COLSS OOS) margin penalty will be applied in Cycle 13 to preserve the limiting dose consequences presented in the FSAR for the inadvertent opening of an atmospheric dump valve (with a concurrent single active failure). Also, setpoints will be adjusted in Cycle 13 to preserve the required margin needed to accommodate the increase in pressurizer spray flow specified.

Revised setpoints have been determined (References 7 and 8) and will be implemented per station procedures prior to the startup of Cycle 13. Also, the requirements of Table 7-4 of Reference 5 (Non-LOCA Safety Analysis External Requirements on the Waterford-3 Groundrules) have been adequately addressed for Cycle 13 and are documented in Reference 53.

Emergency Core Cooling System (ECCS) Performance Analysis

An ECCS performance evaluation was performed for Cycle 13 to demonstrate conformance to the ECCS Acceptance Criteria for Light Water Nuclear Power Reactors (Reference 28). The ECCS performance analysis is comprised of the Large Break Loss of Coolant Accident (LBLOCA), Small Break Loss of Coolant Accident (SBLOCA), and post-LOCA Long Term Cooling (LTC) analyses.

The Cycle 13 ECCS performance evaluation demonstrated that the results of the current analyses for LBLOCA, SBLOCA, and LTC apply to Cycle 13. The break that results in the highest peak cladding temperature (the limiting break), is from the LBLOCA analysis. The results for the limiting break are bounded by the results for the UFSAR Worst Case Analysis and conform to the ECCS acceptance criteria (Peak Cladding Temperature ≤2200°F, Maximum Cladding Oxidation ≤17%, Maximum Core-Wide Cladding Oxidation ≤1%, and maintaining a coolable geometry and long term cooling). The results are applicable for the Peak Linear Heat Generation Rate (PLHGR) reported in the COLR and a licensed core power level (including the power measurement uncertainty) of 3458 MWth.

II. SCREENING

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☒	☐	*See listing in the "Description of Proposed Change" section of this evaluation.*
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☒	☐	*See listing in the "Description of Proposed Change" section of this evaluation.*
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculation Manual[3]	☐	☒	

[1] If "YES," see Section 5.1.4. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

Licensing Basis Documents (References cited are at the end of this evaluation)

Operating License (Operating License, Technical Specifications, NRC Orders)

The Cycle 13 RAR describes and addresses the design, accident analyses, and performance of the Cycle 13 core. A review of the Waterford 3 Operating License determined that the information in the license is not impacted by the changes for Cycle 13.

The Cycle 13 core design and analyses results meet Technical Specification requirements. Safety Limits, Limiting Safety Settings and Limiting Conditions of Operations governing the operation of the Cycle 12 core are bounding for the Cycle 13 core. Therefore, no changes to Technical Specifications are required as a result of reload analysis activities for Cycle 13. Cycle-specific limits for operation of the Cycle 13 core will be located in the Cycle 13 COLR.

NRC Confirmatory Orders and Immediate Orders were reviewed and it was determined that these documents do not address core reload details or other specific information addressed in the Cycle 13 RAR.

LBDs Controlled Under 50.59 (FSAR, TS Bases, TRM, COLR, NRC SERs)

The results of the reload analyses for Cycle 13 make it necessary to update the Waterford 3 FSAR to present current information about Cycle 13. Therefore, a 10CFR50.59 evaluation is required. The FSAR sections, tables, and figures to be changed are given in Section I of this evaluation under "Description of Proposed Changes."

As stated previously, no changes to the Technical Specifications are required as a result of the analyses or evaluations performed for Cycle 13. Likewise, no TS Bases changes or TRM changes were identified to be necessary due to the Cycle 13 reload activities.

The Cycle 13 COLR will provide the specific operational limits for the Cycle 13 core. Technical Specification 6.9.1.11 requires that the COLR be revised at each core reload. The Cycle 13 COLR will be prepared in accordance with NOECP-702 based on the information in Reference 7. This 10CFR50.59 evaluation addresses the COLR changes for Cycle 13.

The results of the Cycle 13 reload analyses as described in the RAR are consistent with the requirements in NRC SERs for Waterford. Changes to COLSS/CPC setpoints and the COLR for Cycle 13 ensure that the core is operated such that analyses requirements are met.

LBDs Controlled Under Other Regulations

Reload analyses inputs and results are not addressed in the QAPM, E-Plan, Fire Protection Program, and the ODCM. Based on this, no changes to these documents are required to support the operation of the Cycle 13.

Test or Experiment

The methods used to design and analyze the Cycle 13 core are approved by the NRC and referenced in the FSAR, Technical Specifications, and the COLR. The Cycle 13 core reload will be implemented in accordance with station procedures and Technical Specifications. Therefore, the Cycle 13 reload does not constitute a test or experiment.

Independent Spent Fuel Storage Installation

Waterford 3 does not have an Independent Spent Fuel Storage Facility.

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LBDS_50_59 (Group)	*Reload(38 hits), safety analysis (174 hits), safety analyses (65 hits), nuclear fuel (19 hits), ejection (44 hits), Region U (1 hit), core height (7 hits), active core (8 hits), batch (31 hits), batches (13 hits), Cycle 11 (10 hits), Cycle 12 (12 hits)*

LBDs/Documents reviewed manually:

Performed a manual review of UFSAR Chapters 4, 6, 7, 9, and 15; Technical Specifications; the COLR; the Operating License; and the Technical Requirements Manual

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☐ Yes
☒ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ☐ No

Name of Security Plan reviewer (print) / Signature / Date

V. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A, above.

A. Check the applicable boxes below. If any of the boxes are checked, a 50.59 Evaluation is not required. If none of the boxes are checked, perform a 50.59 Evaluation in accordance with Section V. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.6.1.1:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

The proposed activity does not adversely affect a method of evaluation that demonstrates intended functions of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed change already exists per Section 5.6.1.2. Reference 50.59 Evaluation # _____ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.6.1.3. Reference: _____

☐ The proposed activity is controlled by another regulation per Section 5.6.1.4.

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10CFR50.59 Review Program Guidelines for guidance.

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

The Waterford 3 Cycle 13 core will contain 125 irradiated assemblies (Region T and Region U fuel), and 92 fresh Region W assemblies. All 49 Region S assemblies and 43 of the 76 Region T assemblies present in the Cycle 12 core will be discharged. All of the 92 Region W assemblies will have Inconel top spacer grids. The elimination of the alumina spacer disks inside the fuel pins and the shortening of the lower end caps, changes which were introduced in Cycle 12 with Region U fuel, will also be employed in Region W. The fuel pin initial U-235 enrichments in the fresh Region W fuel assemblies will range from 4.20 - 4.55 weight percent. These pin enrichments are very similar to the enrichments of 4.17 and 4.57 weight percent in the Region U fuel assemblies that were loaded in Cycle 12.

The Cycle 13 core was designed on the basis of a nominal cycle energy of 514.5 EFPD for a Cycle 12 energy of 524 EFPD. Evaluations have been completed to demonstrate the applicability of the reload analyses for Cycle 12 energies between 514 and 544 EFPD, and for corresponding Cycle 13 energies from 545.1 EFPD down to 528.5 EFPD.

The Waterford 3 Cycle 12 Safety Analysis Groundrules (Reference 2) were used as input into the Cycle 13 reload analyses. Changes to the Groundrules for Cycle 13 relative to Cycle 12 were primarily related to fuel management and core design and were approved in accordance with NOECP-702, Waterford 3 Reload Process.

The Reload Analysis Report (RAR) was prepared by the fuel vendor to document the results of the reload safety analyses. Changes to the Core Operating Limits Supervisory System (COLSS) addressable constants, Core Protection Calculator (CPC) addressable constants, and the Core Operating Limits Report (COLR) have been determined and are necessary to implement the requirements of the Cycle 13 safety analyses. Changes to the Updated Final Safety Analysis Report (UFSAR) are also required to update it with cycle-specific information for Cycle 13.

Reason for proposed Change:

The Waterford 3 reactor must be refueled for Cycle 13 operation and the UFSAR must be updated to report Cycle 13 information. In addition, the COLR and addressable constants in the CPCs and COLSS must be updated to satisfy the requirements of the safety analyses for Cycle 13.

50.59 Evaluation summary and conclusions

This evaluation determined that implementation of the Cycle 13 core as described herein is acceptable. Approved methods were used to analyze the Cycle 13 core design and it was learned that the frequency and consequences of postulated accidents are not impacted by the changes described for Cycle 13. In addition, plant equipment and systems important to safety will not be adversely affected by the changes. The changes to COLSS and CPC addressable constants for Cycle 13 as well as the update to the COLR will preserve the validity of the Waterford safety analyses allowing Waterford 3 to continue to operate in accordance with its design and licensing bases.

B. License Amendment Determination

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 FSAR? ☐ Yes ☒ No

BASIS: (References cited are at the end of this evaluation)

Fuel is not an accident initiator and no impact to any accident initiator occurs due to the Cycle 13 fuel. The Cycle 13 core design is similar to the Cycle 12 design. There were minor changes in fuel management for Cycle 13 and minor modifications in the mechanical design of new fuel for Cycle 13 (Region W fuel) as compared to the fuel design for Cycle 12 (Region U fuel). The Cycle 13 fuel assembly mechanical design is very similar to that for Cycle 12 and it is compatible with existing fuel handling and storage equipment. The Region W fuel assemblies for Cycle 13 have the same structural cage as that previously used at Waterford 3 and employs the Inconel top grid which is used in 32 of the Region U fuel assemblies. All Region W assemblies are capable of withstanding the expected handling loads (FSAR 4.2.1.1 and 4.2.3.1.5). Since no changes to interlocks, mechanical stops, or administrative controls at the plant are required to accommodate the Cycle 13 fuel, the probability of a fuel handling accident (FSAR Section 15.7.3.4) will not be increased.

The probability of erroneous loading of fuel pellets or fuel pins of different enrichment in a fuel assembly or erroneous placement or orientation of fuel assemblies in the core (FSAR Section 15.4.3.1) due to the design change features in Region W assemblies is not increased. Extensive quality controls are used to ensure that fuel components are built correctly. Final core verification (serial number, core location, and orientation check of each fuel assembly) will be performed after the Cycle 13 core is loaded to ensure that the core configuration is consistent with design documentation. Post-refueling startup testing will be performed to verify design predictions and observe core behavior.

Based on the discussion above, the frequency of occurrence of an accident previously evaluated in the FSAR will not be increased due to installation and operation of the Cycle 13 core.

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 FSAR? ☐ Yes ☒ No

BASIS: (References cited are at the end of this evaluation)

The Region W reload fuel assemblies are acceptable for use in Waterford 3 Cycle 13. The nuclear design of Region W fuel was accomplished using NRC-approved analysis methodologies under approved quality assurance programs. The performance of the Region W assemblies is not expected to be significantly different than previous fuel batches. There is acceptable mechanical design margin for the Cycle 13 core containing Region W fuel assemblies and other resident fuel batches. The Region W fuel employs Inconel top spacer grids and assemblies meet all required design and functional requirements. Adequate shoulder gap is predicted for all of the batches of fuel in the Cycle 13 core. There are no fuel mechanical issues that would require Cycle 13 water chemistry to be operated outside the established industry guidelines. 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. In fact, the design objective of the improved assembly design having the Inconel top spacer grid is to reduce grid-to-rod fretting damage at the most vulnerable elevation of the fuel assembly. Evaluations of the improved assembly design demonstrate that it satisfies all design criteria (Reference 27).

The loading of the Cycle 13 fuel assemblies in the core, and subsequent operation will not require

any physical changes to plant equipment or systems. The Region W fuel assemblies are very similar in design to the Region U fuel (Cycle 12 reload) and no impact on the performance or reliability of any plant systems should result from their use.

Incore instrumentation accuracy and response characteristics will not be impacted by installation of Region W fuel. The reduction in fuel pellet stack elevation will change the positions of the incore neutron detectors relative to the active fuel by 0.355 inch, but this will have a negligible impact on the measured power distributions. Similarly, the effect of the fuel elevation change on the excore detector response (and the associated excore Axial Shape Index and Shape Annealing Function) will be insignificant. The peripheral assembly integrated powers generally will be similar to those in Cycle 12 and the neutron flux at the excore detectors in Cycle 13 is predicted to be about 1.6 percent larger than in Cycle 12. Neutron fluence in the reactor vessel and internals remains well within orignial design parameters (including peak fluence in the mid-vessel beltline region) due to low-leakage fuel management implemented early in the life of Waterford 3. Hence, there is no adverse impact on the behavior of materials of the reactor vessel or internals.

The Cycle 13 core design with Region W fuel 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 or require that any systems be operated outside their design limits. The dimensions and placement of the guide tubes in the Region W fuel assemblies are the same as Region U fuel assemblies, and as such, there are no compatibility issues with CEAs. Also, the CEA position measurement system is not being altered in any way to use the Region W fuel.
The active fuel elevation change will result in a small decrease of 0.355 inches in CEA insertion, but the resulting slight reduction in shutdown reactivity worth, and the small increase in negative reactivity insertion time, are within the allowances assumed in the safety analyses. Reducing the bottom elevation of the active fuel is beneficial for ECCS performance during a small-break or large-break LOCA, and for post-LOCA long-term cooling, but the magnitude is negligible for a 0.355-inch elevation change.

Based on the discussion above, there is no characteristic of the Cycle 13 core, with the Region W 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 13 core reload.

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

BASIS: (References cited are at the end of this evaluation)

As a result of the Cycle 13 core reload, there will be no new pathways to the environment created for radioactive material release. The equipment designed to either mitigate the radiological consequences of an accident, or control the release of radioactive material will not be affected.

All LOCA and non-LOCA transients have been evaluated for Cycle 13 (Reference 5) and the results are summarized in Section I of this evaluation. All events were found either to be bounded by the AOR directly or after evaluation when suitable setpoints were established to address necessary margin requirements. These setpoints (COLSS, CPCs, and COLR updates) will be installed prior to Cycle 13 operation.

The characteristics of the Cycle 13 fuel are bounded by the assumptions used in the spent fuel pool rack, temporary fuel storage rack, and fuel carrier criticality analyses. These criteria were not exceeded for the Cycle 13 core so these existing criticality analyses is acceptable for Cycle 13.

The Region W fuel assemblies, with minor mechanical design changes and equipped with the Inconel top grids, have the same envelope, materials, dimensions and structural cage as those previously used at Waterford 3. Adequate shoulder gap is predicted for all of the batches of fuel

used in the Cycle 13 core. The chemical and metallurgical performance of the Region W fuel is expected to be unchanged from the Region U fuel. As such, no change will occur in the radiological release rate/duration, no new release mechanisms are postulated, and no impact will occur to any radiation release barriers.

The characteristics of the Region W fuel will not significantly affect the consequences of a fuel handling accident as currently presented in UFSAR Section 15.7.3.4. The design of the Region W assemblies is the same as for Region U relative to connection of a handling tool to the assemblies. The weight of the Region W assemblies is essentially the same as the Region U assemblies. The assumption of the number of fuel pins that are postulated to fail during the design basis fuel handling accident is still the same and the release inventory assumed in the analysis of the accident is still valid. Therefore, consequences of a design basis fuel handling accident will not increase.

Based on the above, the consequences of accidents previously evaluated in the FSAR will not be increased due to installation and operation of the Cycle 13 core.

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 FSAR? ☐ Yes ☒ No

BASIS:

The Cycle 13 fuel management and design changes require no equipment modifications. The design differences between Region W fuel and Region U fuel are relatively minor and do not impact any SSC important to safety. Important equipment will function in the same manner with the Cycle 13 reload core as with the previous core. The function and duty of the equipment important to safety is not altered. No changes in the assumptions concerning equipment availability or failure modes have been made. Thus, the consequences of a malfunction of equipment important to safety are not increased by the changes associated with the Cycle 13 reload.

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

BASIS:

Installation and operation of the Cycle 13 core does not introduce an accident initiator or potential single failure not already considered in the Updated FSAR. The fuel assemblies comprising the Cycle 13 core (including the new Region W fuel) meet all required design and functional requirements. The nuclear design of Region W fuel was accomplished using NRC-approved analysis methodologies under approved quality assurance programs. The performance of the Region W assemblies is not expected to be significantly different than previous fuel batches. The Cycle 13 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 resulting from the Cycle 13 core reload nor is it necessary to modify the design, function, or operation of any equipment or to install new equipment.

There were no changes in the failure modes of equipment important to safety as a result of the design and analyses associated with the Cycle 13 reload. No initiators of any of the accidents already postulated are impacted by the Cycle 13 reload. Therefore, operation of Waterford 3 with the Cycle 13 core will not cause an accident of a different type than any previously evaluated in the FSAR.

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

BASIS:

As stated previously, the Cycle 13 fuel management and design changes require no equipment modifications. The design differences between Region W fuel and Region U fuel are relatively

minor and do not impact any SSC important to safety. Important equipment will function in the same manner with the Cycle 13 reload core as with the previous core. There are no new modes of failure associated with any of the changes identified previously in Section I of this evaluation for Cycle 13. The new Inconel top grid utilized on some Region U and all Region W assemblies meets all design and functional requirements and has not shown evidence of problems during Cycle 12. No changes due to the Cycle 13 reload analysis will significantly alter the way in which Waterford 3 operates. Cycle-specific setpoints (COLSS, CPCs, and COLR updates) will be installed prior to Cycle 13 operation.

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 13.

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

BASIS:

The Cycle 13 core (including the new Region W fuel assemblies) meet all required design and functional requirements. The design of the Cycle 13 reload was accomplished using NRC-approved analysis methodologies under approved quality assurance programs. The reload design process has considered all required postulated events and found them either to be bounded by the AORs directly or after evaluation when suitable setpoints (COLSS, CPCs, and COLR updates) were established to address necessary margin requirements. In addition, since the nuclear characteristics of the Cycle 13 fuel are bounded by the assumptions used in the applicable criticality analyses, the results of these analyses are also bounding for Cycle 13.

Due to the removal of the alumina spacer disks and the other minor design changes, there will be a small net increase in fuel pin internal void volume. This will result in a decrease in maximum gas pressure and a reduction in predicted fuel temperatures in the safety analyses. Predicted maximum fuel rod internal pressures are less than the critical pressure determined in accordance with Reference 63.

Based on a review of the fuel vendor's reload analysis results, the Updated FSAR, and the Bases of the Waterford 3 Technical Specifications, the design basis limits for the fuel cladding, RCS boundary, and containment will not be exceeded for Cycle 13.

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

BASIS:

In accordance with Technical Specification 6.9.1.11.1, the Cycle 13 core was designed and evaluated using NRC-approved analysis methodology (References 56 through 63) under an approved quality assurance program. For Cycle 13, this methodology included the application of the *No Clad Lift-Off* and erbia burnable absorber methodologies. No new methodologies were required to verify that previous safety analyses (AORs) are applicable to Cycle 13 or to perform the necessary cycle-specific CEA Ejection Fuel Failure analysis. In addition, COLSS and CPC setpoints required for Cycle 13 operation have been determined using the same approved methods as for Cycle 12. Therefore, there has been no deviation from the methods of evaluation described in the FSAR.

References

1. Entergy Letter W3C1-2003-0003, "Cycle 13 Reload Safety Analysis Groundrules", J.B. Holman to J.M. Betancourt, February 10, 2003
2. Calculation EC-S99-005, Rev. 1, DRN 03-167, "Safety Analysis Groundrules," February 10, 2003
3. Entergy Letter W3C1-2003-0015, "Cycle 13 Reload Safety Analysis Groundrules (Revised)", J.B. Holman to J.M. Betancourt, July 7, 2003
4. Calculation EC-S99-005, Rev. 1, DRN 03-919, "Safety Analysis Groundrules," July 7, 2003
5. Westinghouse Letter NF-03-W-WTFD-33, "Waterford 3, Cycle 13 Final Reload Analysis Report (RAR)", J. M. Betancourt to Dennis E. Barr, September 13, 2003
6. Westinghouse Letter NF-03-W-WTFD-28, "Recommended SAR Updates for Waterford 3, Cycle 13", J. M. Betancourt to Dennis E. Barr, August 5, 2003
7. Westinghouse Letter NF-03-W-WTFD-42, "Waterford 3, Cycle 13 Core Operating Limit Report (COLR)", J. M. Betancourt to Dennis E. Barr, October 16, 2003
8. Westinghouse Letter NF-03-W-WTFD-40, "Waterford 3, Cycle 13 Startup Test and Setpoints Transmittal", J. M. Betancourt to Dennis E. Barr, October 2, 2003
9. Westinghouse Letter NF-03-W-WTFD-36, "Waterford 3, Cycle 13 COLSS Database and Test Information", J. M. Betancourt to Dennis E. Barr, September 18, 2003
10. Westinghouse Letter NF-03-W-WTFD-29, "Waterford 3, Cycle 12 Extension Report", J. M. Betancourt to Dennis E. Barr, August 28, 2003
11. Westinghouse Letter NF-03-W-WTFD-27, "Waterford 3, Cycle 13 Draft Reload Analysis Report (RAR)", J. M. Betancourt to Dennis E. Barr, August 5, 2003
12. Entergy Letter CEXO 2003-00371, "Waterford 3 Cycle 13 Draft RAR Comments", D.E. Barr to J.M. Betancourt, August 21, 2003
13. Westinghouse Letter NF-03-W-WTFD-31, "WSES-3, Cycle 13 Full Core Load Map and Refueling Criticality Considerations", J. M. Betancourt to Dennis E. Barr, August 20, 2003
14. Westinghouse Letter NF-03-W-WTFD-38, "WSES-3, Cycle 13 Final RBC", J. M. Betancourt to Dennis E. Barr, September 30, 2003
15. Entergy Letter W3C1-03-0032, "Refuel 12 Boron Concentration Requirement (Mode 6)", C.L. Alday to K.T. Walsh, October 21, 2003
16. Westinghouse Letter NF-03-W-WTFD-8, "Waterford 3, Cycle 13 Calculation Plan", J. M. Betancourt to Dennis E. Barr, February 21, 2003
17. Westinghouse Letter NF-03-W-WTFD-41, "Waterford 3, Cycle 13 Startup Test Predictions", J. M. Betancourt to Dennis E. Barr, October 14, 2003
18. Westinghouse Letter L-2002-049, "WSES-3 Cycle 13 COLSS Interim Thermal Margin Assessment", J. M. Betancourt to Dennis E. Barr, November 1, 2002
19. Westinghouse Letter NF-03-W-WTFD-3, "WSES-3 Cycle 13 COLSS Interim LHR Thermal Margin Assessment Using Time in Cycle Uncertainty Values", J. M. Betancourt to Dennis E. Barr, January 22, 2003
20. Core Operating Limits Report (COLR), Cycle 12 Rev. 0
21. Condition Report CR-WF3-2003-00631 (Maximum Pressurizer Spray Flow), March 10, 2003
22. Westinghouse Letter LTR-NEM-03-354, "Offer for Westinghouse Evaluation of Increased Pressurizer Spray Flow for Waterford 3 Cycle 13", Roger C. Fagan to Jerry Holman, April 17, 2003
23. Entergy Operations, Inc. Contract Order No. 10033547 dated May 1, 2003
24. Waterford 3 Letter W3C1-2003-0001, "Nominal RCS Flow Rate for Groundrules", January 16, 2003
25. Condition Report CR-WF3-2003-00120 (Nominal RCS Flow Rate), January 17, 2003

26. CEN-386-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," ABB Combustion Engineering Nuclear Fuel, August 1992.
27. L-2002-010, "WSES-3, Batch U Fuel Assembly Design with Top Inconel Grid Technical Evaluation Report, Revision 01," February 5, 2002
28. Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
29. Waterford Steam Electric Station, Unit No. 3, Final Safety Analysis Report (FSAR), Facility Operating License Number NPF-38, Docket No. 50-382, through Revision 12C.
30. Technical Specifications, Waterford Steam Electric Station, Unit No. 3, Docket No. 50-382, Appendix "A" to License No. 38, through Amendment 191.
31. Regulatory Guide 1.7 Revision 2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," November 1978.
32. 10CFR50.44
33. 3-A-8 Calc. 02 Rev. 4, "Waterford 3 Combustible Gas," November 1983.
34. ENEAD-01-NP-A Rev. 0, "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," December 1993.
35. ENEAD-02-NP-A Rev. 0, "Verification of CECOR Coefficient Methodology for Application to Pressurized Water Reactors of the Entergy System," September 1994.
36. CR-WF3-2000-1038, "CE/Echelon Methodology," September 6, 2000.
37. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," March 23, 1972.
38. NEAD SR 2003/063.R0, "WSES-3 Fuel Mechanical Input to Cycle 13 10CFR50.59 Reload Evaluation", October 22, 2003
39. Entergy Letter CEO 2003-00208, "WSES-3 Fuel Mechanical Input to Cycle 13 10CFR50.59 Reload Evaluation", D.E. Barr to P.M. Melancon, October 22, 2003
40. Entergy Letter CEO-2003/0184, "Input for Cycle-13 Reload 10CFR50.59 Evaluation", Robert B. Lang to Jerry B. Holman, September 29, 2003
41. NEAD SR 2003/057.R0, "Waterford-3 Cycle-13 Input For Reload 10CFR50.59 Review", September 30, 2003
42. Westinghouse Letter L-2002-052, "Manufacturing Order and Supporting Design Information for WSES-3, Region W", J. M. Betancourt to Dennis E. Barr, November 18, 2002
43. Westinghouse Letter NF-03-W-WTFD-18, "Supplement 01 to Manufacturing Order and Supporting Design Information for WSES-3, Region W", J. M. Betancourt to Dennis E. Barr, April 4, 2003.
44. Westinghouse Letter NF-03-W-WTFD-22, "Pellet Stack Drawing – Pellet Stack Configuration Addition (Technical Change Request)", J. M. Betancourt to Dennis E. Barr, May 2, 2003.
45. Entergy Letter CEO 2003-00026, "Waterford 3 Cycle 13 Core Design Concurrence", D.E. Barr to File, January 23, 2003
46. Entergy Letter CEXO 2003-00302, "Waterford 3 Cycle 13 Core Design", D.E. Barr to J.M. Betancourt, January 23, 2003
47. Entergy Letter CEXO 2003-00473, "WSES-3 Cycle 13 Revised Final Energy Utilization Plan", D.E. Barr to J.M. Betancourt, January 23, 2003
48. Westinghouse Letter L-2002-041, "WSES-3 Cycle 13 Final Core Design Report (FCDR)", J. M. Betancourt to Dennis E. Barr, October 16, 2002
49. Westinghouse Letter NF-03-W-WTFD-1, "WSES-3 Cycle 13 Final Core Design Report (FCDR) for Revised FEUP", J. M. Betancourt to Dennis E. Barr, January 16, 2003

50. Entergy Letter W3C1-2002-0104, "License Amendment Request NPF-38-245, Use of CEN-372-P-A", J.E. Venable to U.S. Nuclear Regulatory Commission, December 16, 2002
51. *Condition Report CR-WF3-2002-00443 (Fuel Rod Internal Pressure)*
52. NOECP-702, Revision 3 Change 2, "Waterford 3 Reload Process", January 14, 2003
53. Entergy Letter W3C1-2003-0033, "Disposition of Cycle 13 Reload Analysis Report External Contingencies", P.M. Melancon to J.B. Holman, October 24, 2003
54. Westinghouse Letter NF-03-W-WTFD-45, "Waterford 3, Cycle 13 Startup Test and Setpoints Transmittal - Corrections", J. M. Betancourt to Dennis E. Barr, October 24, 2003
55. License Amendment 191, "Waterford Steam Electric Station, Unit 3 – Issuance of Amendment RE: Addition of Topical Report Entitled "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A, to the List of Analytical Methods in Technical Specification 6.9.1.11.1 to Determine the Core Operating Limits (TAC No. MB6964), October 31, 2003
56. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983; and "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorber," CENPD-275-P-A, May 1988. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).
57. "C-E Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, January 1976. (Methodology for Specification 3.1.3.6 for Regulating and Group P CEA Insertion Limits and 3.2.3 for Azimuthal Power Tilt).
58. "Modified Statistical Combination of Uncertainties" CEN-356(V)-P-A, May 1988. (Methodology for Specification 3.2.4 for DNBR Margin and 3.2.7 for ASI).
59. "Calculative Methods for the CE Large Break LOCA Evaluation Model For The Analysis of C-E and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985. (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt and 3.2.7 for ASI).
60. "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," CENPD-137-P, August 1974: Supplement 2-P-A, April 1998. (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt and 3.2.7 for ASI).
61. "CESEC - Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", CENPD-107, December 1981. (Methodology for Specification 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.1 for Movable Control Assemblies – CEA Position, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, and 3.2.3 for Azimuthal Power Tilt).
62. "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," ENEAD-01-P, Revision 0, December 21, 1993. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and Group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (calculation of CBC & IBW), and 3.9.1 Boron Concentration).
63. "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A, May 1990. (Methodology for Specification 3.2.1, Linear Heat Rate).

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 Steam Electric Station

Document Reviewed: ER-W3-2003-0117-000 **Change/Rev.:** 0/0

System Designator(s)/Description: *Steam Generators, COLSS, COLR*

Description of Proposed Change

This change is to the Core Operating Limits Report (COLR) and the Core Operating Limit Supervisory System (COLSS) to allow operation with greater than 500 but less than 600 plugged tubes in one or both steam generators.

Prior to Cycle 13,the safety analysis for Waterford 3 supported 500 plugged tubes per steam generator. As a result of tube plugging performed during Refuel 12, there will be 571 plugged tubes in Steam Generator #1 and 484 in Steam Generator #2 during Cycle 13. The emergency core cooling system (ECCS) Performance analysis for Cycle 13 includes a restriction to compensate for the impact of exceeding 500 plugged tubes in a steam generator. This restriction requires a reduction of 0.1 kW/ft in the Peak Linear Heat Generation Rate (PLHGR) LCO to be implemented in the COLSS software and in the Cycle 13 COLR for number of plugged tube greater than 500 but less than 600 in either of the Steam Generators. FSAR Section 6.3.3 requires a minor revision to address the revised PLHGR limits for Cycle 13.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: *03-017*)	Sections I, II, III, IV, and VI required

Preparer: *C. L. Alday / Clint L. Alday / EOI/ Nuclear Engineering/ 11/16/03*
Name (print) / Signature / Company / Department / Date

Reviewer: *J. B. Holman / J B Holman / EOI/ Nuclear Engineering/ 11/16/03*
Name (print) / Signature / Company / Department / Date

OSRC *K J Peterson 11/16/03 / 11/16/03*
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.9) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name: **Scope of Assistance:**

II. SCREENING

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # and/or SECTIONS IMPACTED
FSAR	☒	☐	Section 6.3.3.1 (DRN 03-1943); Tables 15-6.13a & b (DRN 03-1964)
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☒	☐	Cycle 13 COLR (Revision 0), Figures 6 & 7
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculation Manual[3]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation? ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. **Basis**

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The Waterford-3 License Research System (LRS) Entergy Fulfind electronic search was utilized for the License Basis Document applicability determination as detailed in the references in Section C below. This change requires a revision to the COLR, a COLSS FLIM constant change, and a minor revision to FSAR Section 6.3.3. Revision 1 to the COLR has been prepared and will be issued following approval of this evaluation. COLSS FLIM Software Change Request has been prepared and will be implemented following approval of this evaluation. DRN 03-1943has been prepared for the change in PLHGR to update the FSAR following approval of this evaluation. The FSAR references the 500 plugged tubes in Tables 15.6-13a and 15.6-13b and DRN-03-1964 has been prepared to update these tables.

This change will not impact any component's ability to perform its intended design function. The Safety Evaluation Report, Technical Specifications and Technical Requirements Manual were reviewed and found unaffected. Technical Specifications describes the LCOs and surveillances associated with PLHGR, however this change has no affect on those activities. The Waterford-3 facility currently has no provisions for dry fuel storage.

The document searches detailed in the following section describe the FSAR Sections and LBDs reviewed for impact by the proposed change.

C. **References**

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:

LBDS_50_59 (Group)

Keywords:

"steam generator tube plugging", "peak linear heat rate", "linear heat generation rate", "12.6", "linear heat", "500"

LBDs/Documents reviewed manually:*FSAR Section 6.3, FSAR Chapter 15, Technical Specification 3.2.1,Technical Specification 3.4.4, COLR Section 3.2.1, COLR Figures 6 & 7*

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☐ Yes
☒ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ☐ No

Not Applicable

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

This change is to the Core Operating Limits Report (COLR) and the Core Operating Limit Supervisory System (COLSS) to allow operation with greater than 500 but less than 600 plugged tubes in one or both steam generators.

Prior to Cycle 13,the safety analysis for Waterford 3supported 500 plugged tubes per steam generator. As a result of tube plugging performed during Refuel 12, there will be 571 plugged tubes in Steam Generator #1 and 484 in Steam Generator #2 during Cycle 13. The ECCS performance analysis for Cycle 13 includes a restriction to compensate for the impact of exceeding 500 plugged tubes in a steam generator. This restriction requires a reduction of 0.1 kW/ft in the Peak Linear Heat Generation Rate (PLHGR) LCO to be implemented in the COLSS software and in the Cycle 13 COLR for number of plugged tube greater than 500 but less than 600 in either of the Steam Generators. FSAR Section 6.3.3 requires a minor revision to address the revised PLHGR limits for Cycle 13.

Technical Specification 3/4.2.1, Linear Heat Rate, ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200 °F. This Technical Specification applies in Mode 1 above 20% of RATED THERMAL POWER. COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limit specified in the COLR is not exceeded. The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the control room in the event that the core power exceeds the core power operating limit. A change to COLSS FLIM constants will be implemented to update the peak linear heat rate limits as defined above for COLSS in service (COLR Figure 6) and COLSS out of service (COLR Figure 7) to ensure that the annunciator is activated at the correct value of PLHGR. Therefore, the limiting condition for operation on peak linear heat rate will still be satisfied above 20% rated thermal power as required by Technical Specification 3/4.2.1. FSAR Section 6.3.3 presents the performance evaluation of the Emergency Core Cooling System. The specific range for the peak linear heat generation rate for Cycle 13 will be added to this section. The FSAR references the 500 plugged tubes in Tables 15.6-13a and 15.6-13b and will be updated to reflect the new values.

Reason for proposed Change:

Refuel 12 steam generator inspections and tube plugging resulted in plugging greater than 500 but less than 600 tubes in Steam Generator #1. This exceeds the previous limit of 500 plugged tubes and requires an additional restriction on the PLHGR in the COLR and COLSS as described in Westinghouse Letter CWTR3-03-139 dated October 3, 2003.

50.59 Evaluation summary and conclusions:

Reducing the PLHGR by 0.1kW/ft over the core operating inlet temperature range will conservatively account for greater than 500 but less than 600 plugged tubes in SG #1 as described in Westinghouse Letter CWTR3-03-139 dated October 3, 2003.

This activity does not adversely affect the design function of an SSC as described in the FSAR. COLSS continuously monitors the core power distribution and calculates a core power operating limit corresponding to the allowable peak linear heat rate, this function will be not be changed. COLSS provides an automatic alarm when PLHGR exceeds its allowable limit, reducing the allowable PLHGR causes this alarm to annunciate at a lower, more conservative value. Therefore this change does not adversely affect the design function of an SSC, i.e., COLSS, as described in the FSAR.

In summary, the reduction of allowed PLHGR in the COLR and COLSS will not affect their design function. The change will not affect the probability or consequences of an accident described in the FSAR, nor introduce the possibility of a new type of accident. The change will not affect the probability or consequences of a malfunction, nor create the possibility of a malfunction with a different result. Finally, the change does not represent a departure from a method of evaluation described in the FSAR. Therefore, the proposed change does not involve an unreviewed safety question under 10CFR50.59.

B. License Amendment Determination

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 FSAR? ☐ Yes ☒ No

   BASIS:

   This change does not adversely affect any structure, system, or component that is an accident initiator in the FSAR. The tube plugs are installed according to approved procedures that reduce the probability of a steam generator tube rupture. The number of SG tubes plugged has no impact on the likelihood of an accident. COLSS only provides assistance to the operators in maintaining the thermal margin-related Technical Specification LCOs. It does not initiate any equipment actions that could cause any accident.

   In summary, the proposed change does not result in more than a minimal increase in 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 FSAR? ☐ Yes ☒ No

   BASIS:

   The proposed activity does not adversely affect the design function of an SSC as described in the FSAR. The tube plugs are installed in accordance with approved procedures in a manner that minimizes the potential for failure of the plug itself. The plug is installed in order to reduce the likelihood of a steam generator tube rupture.

   COLSS FLIM constants will be updated to reflect the revised PLHGR limits. This will ensure that COLSS continues to accurately represent the core and perform its intended function. The lower allowed PLHGR will preserve sufficient margin to safely operate the reactor at full power. Since COLSS only provides assistance to the operators in maintaining the thermal margin-related Technical Specification LCOs, a reduction in allowed PLHGR does not increase the likelihood of a malfunction of COLSS (it still conservatively performs its design function).

   In conclusion, a reduction in the allowed PLHGR will not cause more than a minimal increase in the likelihood of malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

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

BASIS:

As described in Westinghouse Letter CWTR3-03-139 dated October 3, 2003, exceeding 500 plugged tubes in a steam generator for Cycle 13 has been evaluated. The increase in SG tube plugging will have the effect of decreasing both the total primary system fluid mass and the active steam generator heat transfer area. This has the potential to adversely impact several of the safety analysis transients. All Chapter 15 accident analyses were reviewed for impact due to the increase in SG plugging in WSES-3 beyond 500 per SG as stated in Reference 2. Six non-LOCA events were determined to be potentially impacted by the increase in SG tube plugs above 500. The Loss Of Condenser Vacuum and Feedwater Line Break were examined for potential impact on peak RCS pressure and found to have no impact. The potential thermal margin impact on the Subcritical Control Element Assembly Withdrawal was quantified and no compensatory restrictions are required since the available margin for this event exceeds the margin impact due to tube plugging. The Boron Dilution event for time to criticality was found to be acceptable when Cycle 13 specific values of critical boron concentration and inverse boron worth are used. The Return To Power Steam Line Break was specifically analyzed for WSES-3 with 1000 SG tubes plugged and its results were incorporated into the Cycle 13 Reload Analysis. The Inadvertent Opening of a Steam Generator Atmospheric Dump Valve was determined to not be impacted.

It was also found that RCS flow coastdown data remains valid for Cycle 13 with as many as 700 plugged tubes per steam generator.

Both large break and small break LOCA, ECCS Performance analyses, were evaluated for the impact of SG tube plugging in excess of 500 per SG. Sensitivity analyses show that for up to 700 tubes plugged, the peak clad temperature for SBLOCA remains less than that for the LBLOCA. However, a decrease in the allowed peak linear heat generation rate is required to maintain acceptable results for the LBLOCA analysis. A reduction of 0.1 kw/ft for each 100 tubes plugged in excess of 500 per SG is required, up to a maximum of 700 tubes plugged per SG. This reduction will be implemented via the change to COLSS and the COLR.

Therefore, with the changes to COLSS and the COLR, there will not be more than a minimal increase in consequences to any accident 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 FSAR? ☐ Yes ☒ No

BASIS:

The results of installing additional SG tube plugs do not impact the consequences of a failure of a steam generator tube or the plug. Technical Specifications contain limits on the primary to secondary leakage that minimizes the consequences of such a leak. Those limits are not changed. The main effect of the proposed change is to conservatively reduce the thermal margin during power operation by reducing the allowed PLHGR.

The design function of systems affected by the change is maintained, so there is no impact of the change on the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

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

BASIS:

The proposed change does not have the potential for any initiator-related effect and does not create the possibility of an accident of a different type than previously analyzed in the FSAR. Steam generator tube plugs are currently utilized in steam generators and have been previously analyzed. Changing the number does not create a new accident, nor any new systems or interactions. There are no changes to COLSS functionality. The safety analyses for events that are discussed in FSAR Chapter 15 are not affected by the reduction in the PLHGR limit. This change does not adversely affect any structure, system, or component that is an accident initiator in the FSAR. The required COLR changes will be done by changes to COLR Figures 6 and 7, and the COLSS change will be accomplished via an adjustment to the COLSS FLIM terms. The lower allowed PLHGR will preserve sufficient margin to safely operate the reactor at full power. No new accident initiators are created by this change.

This change does not have the potential for any initiator-related effect and does not create the possibility of an accident of a different type than previously analyzed in the FSAR.

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

BASIS

The proposed change does not involve a malfunction of a system important to safety. All design functions of the structures, system, or components potentially affected by the change continue to be met. Therefore, the proposed change can not create a possibility for a malfunction with a different result than any previously evaluated in the FSAR.

The steam generator tube plugs are installed per approved procedures. Steam generator tube plugs already exist within the Licensing basis, so no malfunction with a different result is created. The proposed activity does not adversely affect the design function of an SSC as described in the FSAR. COLSS FLIM constants will be updated to reflect the revised PLHGR limits. This will ensure that COLSS continues to accurately represent the core and perform its intended function. The lower allowed PLHGR will preserve sufficient margin to safely operate the reactor at full power. Since COLSS only provides assistance to the operators in maintaining the thermal margin-related Technical Specification LCOs, a reduction in allowed PLHGR is not a malfunction of COLSS (it still conservatively performs its design function). As COLSS design functionality is not affected, no new malfunctions are created with a different result than previously evaluated in the FSAR.

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

BASIS:

Additional tube plugs are necessary to maintain existing design basis limits for the fission product barrier, i.e., additional plugs are due to flaw detection in tubes with plugging required to meet design basis limits for flaw size. Plugging of the steam generator tubes is performed to provide assurance that degraded tubes are isolated and the RCS boundary is maintained. The number of tubes plugged does not affect the integrity of this fission product barrier; rather it ensures that tube flaws are isolated. This plugging is done in accordance with approved procedures and work instructions, providing assurance that the integrity of the RCS is maintained. The impact of plugging greater than 500 but less than 600 tubes in Steam Generator #1 is related to COLR and COLSS thermal margin. COLSS will be adjusted via the FLIM constants to ensure that the Technical Specification LCO is maintained and, thus, by definition, the reactor protection system will be able to provide its design function and there will be no impact on accident consequences.

The proposed change does not result in exceeding or altering a design limit for a fission product barrier.

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

BASIS:

Plugging more than 500 and less than 600 tubes in a single steam generator does not represent a change in any method of evaluation. Accident analyses methods used in the FSAR currently assume and account for plugged tubes in the steam generator. Neither the COLR figure changes nor the COLSS constant adjustments represent a change to a methodology described in the FSAR. These adjustments are part of the normal process of incorporating operational limitations into the COLR and COLSS. The use of FLIM constants is part of the current COLSS method of maintaining the Technical Specification LCO; the decrease in allowed PLHGR is simply an application of this existing method to account for plugging greater than 500 but less than 600 tubes in a single steam generator.

Therefore, the proposed changes to the COLR and COLSS to account for additional plugging above 500 tubes do not represent changes to a method of evaluation described in the FSAR.

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 Steam Electric Station 3|V=2

Document Reviewed: Waterford 3 Steam Generator Chemical Cleaning (SGCC) Special Test Instruction (STI-W3-2003-005) **Change/Rev.:** 1

System Designator(s)/Description: Blowdown (BD) System, Controlled Ventilation Area System (CVAS), Condensate Makeup (CMU) System, Sump Pump (SP) System, Main Steam (MS) System, Station Air (SA) System, Shield Building Ventilation (SBV) System, Liquid Waste Management (LWM) System, Reactor Auxiliary Building Normal Ventilation System

Description of Proposed Change

Framatome Advanced Nuclear Power, Inc. (FANP) will be performing the proposed secondary-side chemical cleaning of the Waterford 3 steam generators (SG) at the end of Cycle 12 (the proposed cleaning will occur in Mode 5 – Cold Shutdown). The proposed Special Test Instruction (STI) contains formal instructions for Entergy Operations Inc. (EOI) personnel to control plant operations while FANP personnel perform the chemical cleaning process.

This 10CFR50.59 Evaluation is for the implementation portion of the Steam Generator Chemical Cleaning (SGCC) only. It addresses mainly the effect of the cleaning process on Waterford 3 Technical Specifications and Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses. Note that the effect on the safety related Heating, Ventilating and Air Conditioning (HVAC) systems that contain charcoal is addressed in this evaluation. The connection of the FANP chemical cleaning equipment to plant systems was evaluated separately as part of Procedure PMC-004-008 "Installation Procedure Steam Generator Chemical Cleaning Equipment Installation". ER-W3-2003-0366-02 addresses the impact of the Steam Generator Chemical Cleaning Process on Waterford 3 Permanent Plant Instrumentation. ER-W3-2003-0522-00 addresses the impact of SGCC chemicals on Fuel and Reactor Coolant System (RCS) Components due to leakage (through steam generator tube(s)) from the secondary to primary side during the chemical cleaning. The effects on the steam generator internals and the environmental effects are addressed in separate 10 CFR 50.59 Reports; a "Process" Exemption and an "Environmental" Exemption. These separate reports address the following issues:

- Chemical Process (Affect on steam generator Internals and vapor space corrosion – see "Process" Exemption).
- Control Room habitability (see "Environmental" Exemption).
- Tank Farm Operation and non-FSAR Chapter 15 events (tornado and hurricane - see "Environmental" Exemption).
- Decomposition and process by products (Impact on plant personnel - see "Process" and "Environmental" Exemptions).
- Vendor equipment failure modes and effects analysis (see "Environmental" Exemption).
- Test Results of Waterford 3 specific laboratory testing (see "Process" Exemption).
- Process Corrosion monitoring (see "Process" Exemption).
- Eddy Current tube testing following SGCC (see "Process" Exemption).
- Waste Processing (see "Environmental" Exemption).
- Environmental releases (see "Environmental" Exemption).

The Steam Generator Chemical Cleaning process is conducted in three major parts; iron removal (iron step) followed by SG drying followed by copper/lead removal (copper step). Several iron removal steps (removes magnetite deposits) and several copper/lead steps (removes copper and lead deposits) will be performed for each steam generator during the cleaning process. The actual number of iron and copper steps will be determined during the cleaning process based upon sample results. The iron and copper/lead steps are performed on each steam generator in parallel. The iron step, aimed at bulk deposit dissolution, is applied in a modified fill/soak/drain mode with nitrogen sparging for additional solution mixing. Following the iron step, the steam generators will be dried with an air sparge for approximately 24 hours in preparation for the copper step. During the drying step air is blown through the steam generators and out the Atmospheric Dump Valves (ADVs). The humidity of the air exiting the ADVs is monitored with humidity probes placed in the exit plume area of the ADVs. The copper step will consist of a full bundle application with air sparging for mixing. Hydrogen peroxide will be added during the copper solvent injection to provide additional oxygen for the oxidation and dissolution of copper deposits. After the copper step is complete, two low volume rinses (LVRs) (~3,000 gallons per SG) and a full volume rinse (FVR) (~31,000 gallons per SG) will be performed.

The iron removal steps utilize an iron solvent composed primarily of Diammonium Ethylenediaminetetraacetic Acid (EDTA) to remove the magnetite deposits. The iron solvent is introduced into the steam generator through connections made to the Blowdown System (via blind flanges downstream of valves BD-107A and BD-107B and vendor supplied chemical tanks, pumps and hoses). The Reactor Coolant System (RCS) temperature during the iron steps is maintained at approximately 195° F (range of 190° F to 195° F). The nominal pressure on the secondary side of the steam generators will be maintained from 0-60 psig during the chemical cleaning. The maximum secondary side pressure will not exceed 70 psig (per STI-W3-2003-005). The initial solvent level injection will be to 228" above the tubesheet (~12,100 gallons). The solvent will remain at this level for 4-8 hours depending upon the dissolution rate of the iron deposits. After the initial fill to 228" is complete, nitrogen sparging will be implemented via the blowdown line connection at a rate of approximately 300 scfm (acceptable range of 250-400 scfm) for five minutes and then secured for ten minutes. The nitrogen sparging sequence will be repeated throughout the iron step application. A rate of 250-400 scfm, targeting 300 scfm, will provide a turnover of the SG volume in ~2 minutes and provide adequate mixing of the EDTA solution. The atmospheric dump valves (ADV - MS-116A or MS-116B) will remain partially opened during the iron step except as required to pressurize the steam generators for drain down.

Samples will be collected from the steam generators by draining ~2,000 gallons of solvent from the steam generators to the FANP process tanks and collecting a representative sample at the end of the drain sequence and then returning the solvent volume to the SGs after the sample is completed. Iron Step samples will be taken at various intervals starting at 2.5 hours after the start of the iron step and continuing throughout the entire 40 hour iron step. The final 40 hour drain down sample will be a composite sample from both SGs collected at 3 different levels during SG drain.

When there is indication that iron dissolution has begun to slow at the initial fill level or insufficient free EDTA remains for further dissolution effectiveness, the solvent will be drained from the SGs and reinjected to the final iron solvent fill level of 387" above the

tubesheet (~31,000 gallons of solvent). Draining and refilling the steam generators will require approximately 4-5 hours. An additional drain and refill cycle will be performed after ~14 hours of total exposure. Nitrogen sparging will be implemented via the blowdown line connection at a rate of approximately 300 scfm for five minutes and then secured for ten minutes as previously stated. The sparging sequence will be repeated throughout the iron step application. As discussed above, samples will be collected from the individual SGs throughout the iron step application alternating between SGs. After ~36.5 hours of total exposure final drain down of the SGs well be initiated. Drain down will require ~3.5 hours to complete, resulting in a total qualified exposure time of 40 hours at the SG tubesheet. Following completion of the draindown, the plant will cool the RCS to a temperature between 145°F and 155°F, targeting ~150°F.

Following draindown of the iron solvent and cooldown of the RCS to 150°F, an approximate 24-hour drying step will be implemented by sparging air at a rate of 650-1000 scfm per SG. Following the drying step, the RCS will be cooled down further to a temperature of about 110°F, if achievable. A minimum cooldown temperature of ~140°F is required for the copper step application.

Once the steam generator temperature is stabilized, injection of the ~6,000 gallons of copper solvent (composed primarily of Ethylenediamine or EDA), hydrogen peroxide and Ammonium Carbonate (5,800 gallons of copper solvent, 200 gallons of hydrogen peroxide and 18 grams/liter of ammonium carbonate) will be initiated and followed by an additional 5,600 gallons of copper solvent. After the copper solvent only injection is complete, approximately 400 gallons of hydrogen peroxide will be added (spiked), bringing the SG level up to 228" above the tubesheet (~12,100 gallons). The addition of the hydrogen peroxide will be immediately followed by air sparging at a rate of 650-1000 scfm per SG for ~15 minutes before proceeding to the next copper solvent injection.

After completing the sparging sequence, the remaining copper solvent (~18,000 gallons) will be injected into the SGs followed by another hydrogen peroxide injection of ~1,000 gallons. Additional hydrogen peroxide will again be added (spiked) to the SG. At this time, the SG levels will be at ~387" above the tubesheet (~31,000 gallons). Air sparging at 650-1000 scfm will be implemented for ~30 minutes following completion of the final hydrogen peroxide injection prior to collecting the first sample from SG#1 at full volume. Approximately 57 gallons of diammonium EDTA will be added to each SG with the refill sequence of the first sample sequence performed in each SG after reaching the final fill level ~387" above the tubesheet to achieve a desired total EDTA concentration of ~0.8 g/L.

Copper Step samples will be taken at various intervals starting at 2.0 hours after the start of the copper step and continuing throughout the entire 48 hour copper step. The final 48 hour drain down sample will be a composite sample from both steam generators collected at three different steam generator levels during draindown.

Both steam generators are then rinsed. This completes the chemical cleaning process. The entire steam generator chemical cleaning process is projected to take approximately five days.

See Section VI below for a complete description of the FANP equipment and plant connections required for the SGCC.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: 03-035	Sections I, II, III, IV, and VI required

Preparer: R.T. Finch / R. T. Finch / / EOI / Design Engr. / 10-21-03
Name (print) / Signature / Company / Department / Date

Reviewer: C.E. DeDeaux / C E DeDeaux / / EOI / Licensing / 10-21-03
Name (print) / Signature / Company / Department / Date

OSRC [signature] 10-21-03
Chairman's Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.9) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
1. Ron Stanley	1. SGCC "Process" Assistance.
2. Greg Hood	2. Environmental Review per Procedure EV-115.
3. T.C. Su	3. Equipment Qualification Assistance
4. Bob Varrin	4. Equipment Qualification Assistance

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☒	☐	No revision required.
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculation Manual[3]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☒ Yes ☐ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

[1] If "YES," see Section 5.1.4. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The plant systems which will be used for the proposed Steam Generator Chemical Cleaning are Blowdown, Controlled Ventilation Area, Condensate Makeup, Sump Pump, Main Steam and Station Air. These systems are described in the FSAR and drawings which show the specific details of the plant connection points are included in the FSAR by reference. Since the proposed SGCC has the potential to impact these plant systems Item 1 (FSAR) above was checked "Yes" and an Evaluation has been performed. Also, since the proposed SGCC is not described in the FSAR and Special Test Instruction has not been previously approved Item 2 above was checked "Yes".

Waterford 3 does not currently have an Independent Spent Fuel Storage Facility so Item 3 above was checked "N/A"

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LBDS_50_59	"steam generator chemical cleaning", "chemical cleaning", "blowdown", "CVAS", "Controlled Ventilation Area System", "Condensate Makeup", "Condensate", "Sump Pump", "Atmospheric Dump Valve", "Station Air"

LBDs/Documents reviewed manually:

UFSAR Sections 2.4.2.3, "Effects of Local Intense Precipitation"; 6.5.1.2.1.2, "Controlled Ventilation Area System"; 9.2.6, "Condensate Storage Facilities"; 10.3, "Main Steam Supply System"; 10.4.8, "Steam Generator Blowdown System"; 11.2, "Liquid Waste Management System". Technical Specification Section 3/4.7.7, "Controlled Ventilation Area System". 50.59 Report for ER-W3-2003-0366-005, "Evaluate Radiological and Environmental Controls During Construction Period of the Steam Generator Chemical Cleaning Process". 10CFR50.59 Evaluation No. 00-051, "50.59 Evaluation, Steam Generator Chemical Cleaning – Implementation". FANP Dwg. Nos. 6026845A, "Waterford 3 2003 Chemical Cleaning Sequence Control Procedure"; 6027358A, "Waterford 3 2003 Chemical Cleaning Process Control Procedure"; 51-5030944-00, "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report". Dominion Engineering, Inc. Letter No. L-4160-01-2, "Material and Geometry Review – Waterford 3 Steam Generator Chemical Cleaning", September 15, 2003.

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☒ Yes
☐ No

If "Yes," list the required changes.

1. 10CFR50.59 "Process" Exemption based on Evaluation No. 00-052.
2. 10CFR50.59 "Environmental" Exemption based on Evaluation No. 00-053.
3. Engineering Request No. ER-W3-2003-0366-02, Impact on Permanent Plant Instrumentation.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☒	☐	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☒	☐	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☒	☐	Involve the installation or modification of a stationary or mobile tank?
14.	☒	☐	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ☐ No

N/A

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

The Combustion Engineering 3410 Series steam generators (SGs) at Waterford 3 (W3) were chemically cleaned during the period of October 14 to October 19, 2000 by Framatome / Siemens. This secondary side cleaning was accomplished through application of the Siemens High Temperature Chemical Cleaning process. The process included both iron and copper removal steps. The W3 application was the first performance of the Siemens process in the United States.

The chemical cleaning and subsequent sludge lancing were successful in removing a significant amount of iron and deposit from the steam generators. A total of approximately 9,650 lbs of magnetite was removed from the SGs via chemical cleaning. This value includes both deposit dissolution and corrosion iron. Approximately 3,710 lbs of wet sludge was removed via sludge lancing following the chemical cleaning and another ~900 lbs (~300 lbs from SG #1 and ~600 lbs from SG #2) via sludge lancing following a post-cleaning upper bundle flush (UBF). Although a significant amount of deposits were removed during the 2000 chemical cleaning, residual deposits, specifically copper, need to be removed from the W3 SGs.

The proposed change is to implement a Framatome Advanced Nuclear Power, Inc. (FANP) low temperature (Mode 5) steam generator chemical cleaning during Refuel 12 to remove the residual steam generator deposits. An iron and copper removal step will be implemented to pursue optimal removal of targeted contaminants, copper and lead, within the allowable outage schedule. Removal of the copper is expected to improve the steam generator inspection eddy current signals and potentially improve W3's probability of detection (POD). In addition, the removal of the residual deposits may potentially correct the pressure loss across the steam generators.

The objectives of the cleaning are to remove the remaining available iron deposits present in the SGs with a low temperature (Mode 5) iron removal application to minimize the corrosion impact to the SG internals, and remove as much of the residual copper in the SGs as possible, especially in the crevice regions at the tube and eggcrate support intersection. The primary target for removal of the residual copper is based on a deposit characterization of ~3% in the deposit scale and ~10% in the loose powder deposit. Deposit amounts have been estimated at a nominal loading of 1,500-2,000 pounds of magnetite per steam generator with a maximum worst case loading of up to 3,000 lbs of magnetite per steam generator.

A summary of the steam generator chemical cleaning (SGCC) process is as follows:

1. Iron Steps (removes magnetite deposits)

   a. Performed when the Reactor Coolant System (RCS) is at approximately 190°F to 195°F (Mode 5).
   b. Both steam generators are drained to plant systems.
   c. Approximately 12,100 gallons of iron solvent, composed primarily of Diammonium Ethylenediaminetetraacetic Acid (EDTA) is injected (first injection) into each steam generator at a rate of approximately 150 gallons / minute (GPM) per steam generator. The flow rate can range from 65 GPM to 150 GPM per SG. The solution is injected from the FANP tank farm via the blowdown system (via blind flange connections downstream of valves BD-107A and BD-107B). The Atmosphere Dump Valves (ADVs) (MS-116A and MS-116B) are opened during the solvent injection to vent the steam generators.
   d. The SGs are sparged with nitrogen, supplied from the FANP tank farm area through the blowdown connections, for five minutes on sparge and ten minutes off sparge. The

sparge cycle continually repeats and ensures that complete mixing of the cleaning solution occurs in the steam generators.

e. Every several hours samples are drawn from the steam generators by draining approximately 2,000 of iron solvent to the FANP tank farm (the solvent is reinjected after sampling). Based upon the sample results the steam generators will be completely drained to the FANP tank farm approximately 4-8 hours after the initial iron solvent injection. The draining is accomplished by closing the ADVs and pressurizing the steam generators with nitrogen, supplied from tanks located adjacent to the FANP tank farm and injected through the blowdown connections, to a pressure of approximately 60 psig.

f. Approximately 31,000 gallons of iron solvent is injected (second injection) into the steam generators. The level will be approx. 387" above the steam generator tubesheet. As during the first injection, nitrogen venting will be continuous and sampling will occur periodically.

g. The sample results are again used to determine when the iron solvent will be completely drained from the steam generators.

h. Approximately 31,000 gallons of iron solvent is injected (third injection) into the steam generators. The level will be approx. 387" above the steam generator tubesheet. As during the first and second injections, nitrogen venting will be continuous and sampling will occur periodically.

i. The sample results are again used to determine when the iron solvent will be completely drained from the steam generators. If the sample results are satisfactory this is the completion of the iron steps.

2. Drying Step

a. The Reactor Coolant System temperature is reduced to approximately 150°F.

b. Air is sparged through each steam generator at a rate of 650-1000 scfm (per steam generator) to dry the steam generators prior to the copper step. The air is introduced into the steam generators, from FANP compressors located adjacent to the FANP tank farm, through the blowdown connections. The ADVs are open during the drying step.

3. Copper / Lead Step (removes copper and lead deposits)

a. Performed when the RCS is at a temperature of approximately 110°F (maximum allowable temperature of approximately 140°F).

b. Approximately 5,800 gallons of copper solvent (primarily composed of Ethylenediamine (EDA)) is injected into each steam generator at a rate of approximately 150 GPM. In parallel, 200 gallons of hydrogen peroxide is injected into each SG, at a rate of approximately 10 GPM. Injection is from the FANP tank farm through the blowdown connections.

c. Another approximately 5,600 gallons of copper solvent is injected into each SG. Following this, approximately 400 gallons of hydrogen peroxide is then injected into each SG.

d. Air is sparged through each steam generator at a rate of 650-1000 scfm for approximately 15 minutes.

e. Approximately 18,000 gallons of copper solvent is injected into each SG followed by another injection of approximately 1,000 gallons of hydrogen peroxide.

f. Air is sparged through each steam generator at a rate of 650-1000 scfm for approximately 30 minutes.

g. Every several hours samples are drawn from the steam generators by draining approximately 2,000 gallons of iron solvent to the FANP tank farm (the solvent is reinjected after sampling).

h. For the first sample/reinject sequence approximately 57 gallons of EDTA will be added to each steam generator during the reinjection.

i. After the first sample sequence, a sparging/pressurization/depressurization sequence will be performed in both SGs. The SGs will be sparged with air at a rate of 650-1000 scfm for approximately 1 hour, during which time the ADVs will be open for a short period and then closed in order to build the pressure in the SGs to approximately 45 psig. Once a pressure of 45 psig is gained, sparging will be secured and the SGs held at approximately 45 psig for 30 minutes. After 30 minutes, the ADVs will be opened and the SGs depressurized. Then the sparging sequence will begin again and continue throughout the duration of the step.

j. A drain and refill sequence may be performed at some time after 24 hours of exposure. For the refill approximately 31,000 gallons of copper solvent will be injected into each SG. Sparging and sampling will be performed following the refill.

k. When the sample results are satisfactory nitrogen is injected to drain the spent solvent solution from the steam generators.

3. Low Volume Rinse

   a. The steam generators and connecting lines are blown down with nitrogen to remove residual chemicals.
   b. Approximately 3,000 gallons of demineralized water is injected into each steam generator and then drained by injecting nitrogen. A minimum of two low volume rinses may be performed. The rinse water is sampled during draining.

4. Final Rinse

   a. Approximately 31,000 gallons of demineralized water is injected into each steam generator for a full volume rinse and then drained by injecting nitrogen. The rinse water is sampled during draining.
   b. The steam generators and lines are blown down with nitrogen.

The entire steam generator chemical cleaning process is estimated to take approximately five days and ten hours. It should be noted that the above described chemical cleaning process may be modified/adjusted slightly, during the cleaning process, according to the chemical samples that are taken and analyzed during the actual cleaning. Any changes will be approved, in writing, by both FANP and Entergy personnel.

Waterford 3 has developed a Special Test Instruction (STI-W3-2003-005) for the Steam Generator Chemical Cleaning (SGCC) Project. The title of the STI is "Waterford 3 Steam Generator Chemical Cleaning Special Test Instruction". The STI addresses Waterford 3 Operations Department responsibilities during the chemical cleaning. Waterford 3 Operations personnel (EOI) will maintain overall control of the plant during the cleaning and will operate all permanent Waterford 3 equipment. FANP personnel will operate all FANP supplied equipment and will control the flow of chemicals, air and nitrogen into the permanent plant systems. FANP personnel will work according to their own procedures, which have been reviewed and approved by EOI. Close and accurate communication between the EOI and FANP personnel is essential to the success of the cleaning project. Training, specific to the chemical cleaning will be conducted for the EOI Operations personnel that will be on duty during the chemical cleaning. FANP personnel will also participate in this training. This will ensure that personnel from both organizations are familiar with the procedures and interface requirements prior to the start of the actual steam generator cleaning.

The FANP procedure for controlling the SGCC process is Drawing No. 6026845A, "Waterford 3 2003 Chemical Cleaning Sequence Control Procedure". The set up of the FANP equipment for the SGCC project is shown on the following drawings:

1. Drawing No. 6026331E, "Waterford 3 2003 Chemical Cleaning Site Layout", 3 sheets.
2. Drawing No. 6026253E, "Waterford 3 2003 Chemical Cleaning P & ID", 3 sheets.

3. Drawing No. 6027153E, "Waterford 3 2003 Waste Processing P & ID", 3 sheets.

The majority of the FANP equipment is associated with liquid waste water storage and treatment. A bermed area approximately 132' x 153' has been constructed in the Plant Controlled Area west of the Service Building (outside of the Protected Area). The following equipment will be stored within the berm:

1. Two – 40,000 gallon iron process/waste tanks
2. Two – 40,000 gallon copper process/waste tanks
3. Five – 25,000 gallon waste storage tanks
4. Two – 20,000 gallon batch release tanks
5. One – 2,500 gallon hydrogen peroxide tank
6. One – Vent Trailer
7. Two – 500 gpm injection pumps.
8. Two – 500 gpm drain pumps
9. Two – Rain pumps
10. Two – 500 gpm tank recirculation pumps
11 One – 40' tall Evaporator for use during waste processing.
12. One – Evaporator Support Trailer containing a 1.6 MW Boiler.
13. Various pipes/hoses used to transfer water/chemicals/air/nitrogen between the FANP equipment/plant.

The following additional equipment will be placed adjacent to the bermed area:

1. One - 20' Cooling Tower Trailer
2. One - 45' Transformer Trailer
3. Two - 40' Parts Trailer
4. One - 40' Chemical Addition Trailer
5. One - 40' Administration Trailer
6. One – 40' Maneuvering Trailer
7. One – Chemical Addition Trailer –Utilized for transfer of all bulk chemicals into the various process tanks during solvent dilution and adjustment
8. One - 40' Crew Trailer
9. Five - 40' Hose Trailer
10. One - 40' Berm Trailer
11. One - 40' Radiation Protection Trailer
12. One – 40' Pipe Trailer
13. One – 40' Waste Processing Trailer
14. One – 20' office Equipment Trailer
15. Two – 20' Berm Parts Trailer
16. Four – Air Compressors
17. One – Nitrogen Trailer
18. Various pipes/hoses used to transfer water/chemicals/air/nitrogen between the FANP equipment/plant

The following FANP equipment will be placed inside the Waterford 3 Protected Area (PA):

1. Two – Drain Pumps
2. One – Vent Tank / Pump Skid
3. Various pipes/hoses used to transfer water/chemicals/air/nitrogen between the FANP equipment/plant

FANP Steam Generator Chemical Cleaning equipment has also been connected to Waterford 3 specific plant structures, systems and components (SSCs) in support of the chemical cleaning of the Waterford 3 Steam Generators. This work has been previously authorized by Procedure PMC-004-008 "Installation Procedure Steam Generator Chemical Cleaning Equipment Installation". The scope of the work includes

adding temporary isolation valves to the Blowdown and Controlled Ventilation Area (CVAS) systems, connecting hoses/piping downstream of manual isolation valves/caps on the Condensate Makeup (CMU) system, routing hoses to the Sump Pump (SP) system and Solidification Building, routing hoses/piping to the vendor supplied chemical cleaning equipment, placing humidity probes in the exit plume area of the Main Steam System Power Operated Atmospheric Dump Valves (ADVs), placing high velocity fans adjacent to the ADVs and connecting valves/hoses/piping to the Station Air System. Scaffolding will be erected in various areas to provide support for the FANP hoses/piping. Trenching will be required in some areas where the FANP hoses/piping cross roadways/security areas. Additional details of the work are provided below:

Blowdown System: To provide interface connections to the BD System for SG chemical cleaning, ER-W3-00-0042-03-00 installed 2" diameter blind flanged branch connections on blowdown lines 5BD4-3 and 5BD4-4, near the Blowdown Flash Tank. Procedure PMC-004-008 has authorized removal of the blind flanges on each line, and installation of temporary flange/pipe assemblies that include 2" diameter root valves. The two 2" lines have been routed a short distance and connect to the single 4" FANP process (injection/drain) line. The 4" process line has been routed through the floor plug opening at elevation -4, down to the -35 elevation and then through the unused wall sleeve at elevation -27 (see CVAS description below). The line has been run through the West side wet and dry cooling tower areas, over the Nuclear Island flood wall at elevation +30 (adjacent to the dry cooling towers), across the yard, trenched below the North side roadway and then run behind the Maintenance Support Building (MSB), below the security fence and out to the FANP tank farm/waste treatment area located in the Owner Controlled Area. Temporary hose/pipe connections will also be made to blowdown piping downstream of valve BD-1253 to enable the draining of SGCC chemicals from the Blowdown Flash Tank.

Condensate Makeup System: Demineralized water is required for initial functional testing, flushing the hoses and equipment during the chemical cleaning process and for waste processing, tank washing, and laboratory use prior to, during and following the chemical cleaning process. The water will be provided from a flanged 3" diameter hose connection installed on an existing 6" diameter blind flanged connection downstream of normally closed manual isolation valve CMU-1222. Valve CMU-1222 is a portable demineralizer hookup connection on the discharge of the Condensate Transfer Pump, and it is located near the Secondary Vacuum Degasifier. The 3" hose has been routed west under the security fence and then to the FANP tank farm/waste reduction area.

Controlled Ventilation Area System: The 4" FANP process line enters the Reactor Building (RB) Wing Area at the unused 6" wall sleeve at elevation -27 adjacent to the West Side Wet Cooling Towers. The wall sleeve is not part of any plant system and is normally blind flanged on both ends. FANP has provided temporary piping assemblies that consist of mating 6" flanges, piping, reducers and 4" root valves which have been installed on each side of the wall penetration.

Sump Pump System: A temporary flow path using 2" hose/piping has been created to allow discharge of treated waste water, rain water, cooling water, and eyewash/shower water, to Dry Cooling Tower (DCT) Sump No. 1, located in DCT A from the FANP Batch Release Tank (BRT). The hose has been routed from the FANP Waste Treatment Area to the +30.00 elevation Nuclear Island Floodwall where a pipe assembly has been installed on top of the flood wall. This pipe supports the hose and is anchored to the top of the flood wall to secure the hose. The hose is routed down the face of an interior wall that separates the individual cooling tower bays, and is attached to the catwalks adjacent to the cooling tower fans. The routing and securing of the hose to the catwalk structural steel and/or handrails ensures that the hose cannot fall or strike safety related cooling tower equipment located in the DCT Area. The hose is terminated at a convenient floor drain and anchored to secure the free end of the hose at the drain.

The 2" line will also have a tee junction adjacent to the dry cooling tower area. The second 2" line runs along the top of the flood wall at elevation +30, over the top of the West Side Access Building, across the top of the Tool Room to the Solidification Building where it is connected to the temporary demineralizer skid.

Main Steam System: FANP Humidity Probes will be placed in the exit plume area of the Power Operated Atmospheric Dump Valves (ADVs). This work will not be performed until the plant reaches Mode 5. High velocity fans will also be positioned, adjacent to the ADVs, to accelerate the fumes that are released during the SGCC past the potential downdraft effects of the containment building.

Station Air System: One inch diameter FANP valves/piping/hoses have been connected to Station Air valves SA-712 and SA-8016. The connection to SA-712 provides air to power the FANP transfer pump located in the Reactor Building -35 elevation and the connection to SA-8016 provides air to power the FANP pump in the protected area outside of the Nuclear Island Floodwall. An air hose has been run over the + 30 elevation Nuclear Island Floodwall to supply air to the pump in the protected area. Except for the initial testing station air will only be used in Mode 5.

This 10 CFR 50.59 Safety Evaluation is for the implementation portion of the Steam Generator Chemical Cleaning (SGCC) only (evaluates Waterford 3 Steam Generator Chemical Cleaning Special Test Instruction - STI-W3-2003-005). It addresses mainly the effect of the cleaning process on Waterford 3 Technical Specifications and Updated Final Safety Analysis Report Chapter 15 accident analyses. The connection of the FANP chemical cleaning equipment to plant systems was evaluated separately as part of Procedure PMC-004-008 "Installation Procedure Steam Generator Chemical Cleaning Equipment Installation". ER-W3-2003-0366-02 addresses the impact of the Steam Generator Chemical Cleaning Process on Waterford 3 Permanent Plant Instrumentation. ER-W3-2003-0522-00 addresses the impact of SGCC chemicals on Fuel and Reactor Coolant System Components due to leakage [through steam generator tube(s)] from the secondary to primary side during the chemical cleaning. The effects on the steam generator and the environmental effects are addressed in separate 10 CFR 50.59 Reports; a "Process" Exemption and an "Environmental" Exemption. The separate reports address the following issues:

1. Chemical Process (see "Process" Exemption).
3. Control Room habitability (see "Environmental" Exemption).
4. Tank Farm Operation and non-FSAR Chapter 15 events (tornado, flooding and hurricane - see "Environmental" Exemption).
5. Decomposition and process by products (chemical fume impact on plant personnel - see "Process" and "Environmental" Exemptions).
6. Vendor equipment failure modes and effects analysis (see "Environmental" Exemption).
7. Test Results of Waterford 3 specific laboratory testing (see "Process" Exemption).
8. Process Corrosion monitoring (see "Process" Exemption).
9. Eddy Current tube testing following SGCC (see "Process" Exemption).
10. Waste Processing (see "Environmental" Exemption).
11. Environmental releases (see "Environmental" Exemption).

Reason for proposed Change:

The Combustion Engineering 3410 Series steam generators at Waterford 3 (W3) were chemically cleaned during the period of October 14 to October 19, 2000 by Framatome ANP, Inc. This secondary side cleaning was accomplished through application of the Siemens High Temperature Chemical Cleaning process. The process included both iron and copper removal steps. The W3 application was the first performance of the Siemens process in the United States.

The chemical cleaning and subsequent sludge lancing were successful in removing a significant amount of iron and deposit from the steam generators (SGs). A total of approximately 9,650 lbs of magnetite were removed from the SGs via chemical cleaning. This value includes both deposit dissolution and corrosion iron. Approximately 3,710 lbs of wet sludge were removed via sludge lancing following the chemical cleaning and another ~900 lbs (~300 lbs from SG #1 and ~600 lbs from SG #2) via sludge lancing following a post-cleaning upper bundle flush (UBF). Although a significant amount of deposits were removed during the 2000 chemical cleaning, residual deposits, specifically copper, need to be removed from the W3 SGs. Removal of the copper is expected to improve the steam generator tube inspection eddy current signals and potentially improve the probability of detection (POD) of tube flaws. In addition, the removal of the residual deposits may potentially correct the pressure loss across the steam generators.

The objectives of the cleaning are to remove the remaining available iron deposits present in the SGs and remove as much of the residual copper in the SGs as possible, especially in the crevice regions at the tube and eggcrate support intersection. The primary target for removal of the residual copper is based on a deposit characterization of ~3% in the deposit scale and ~10% in the loose powder deposit. Deposit amounts have been estimated at a nominal loading of 1,500-2,000 pounds of magnetite per steam generator with a maximum worst case loading of up to 3,000 lbs of magnetite per steam generator.

50.59 Evaluation summary and conclusions

The proposed change does not cause the parent systems to be operated outside of their design or test limits, negatively affect any system interfaces or result in an increase in challenges to safety systems or systems important to safety. The proposed activity does not result in a change from one frequency class to a more frequent class or an increase in frequency within a given class. This evaluation concludes that neither the actual chemical cleaning, or the temporary system connections associated with the chemical cleaning, will degrade the integrity or performance of the steam generators, the connected instrumentation, or the affected systems. All physical changes are temporary, and there are no new permanent system interactions created. This change does not require any Technical Specification changes. An Environmental Impact Evaluation was required. As stated above the Process effects internal to the steam generator and the Environmental effects of the proposed Steam Generator Chemical Cleaning are addressed in separate 10 CFR 50.59 Reports (see the 10 CFR 50.59 "Process" Exemption and 10 CFR 50.59 "Environmental" Exemption).

B. License Amendment Determination

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 FSAR? ☐ Yes ☒ No

BASIS:

The probability of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR) will not be increased due to the implementation of the Steam Generator Chemical Cleaning process. The probability of a rupture of a steam generator tube should actually be decreased by the SGCC due to the removal of deposits that can induce corrosion of the steam generator tubes, tubesheets and other internal components. The purpose of the cleaning process is to enable the steam generators to reach their forty-year design life. The chemical cleaning process and the implementation of this process are evaluated below with respect to the Waterford 3 Technical Specifications and to the Updated Final Safety Analysis Report Chapter 15 analyses. To fully answer this question the Technical Specification compliance issues will be addressed first, then the UFSAR Chapter 15 analyses will be addressed.

During implementation of the Steam Generator Chemical Cleaning (SGCC) process Waterford 3 will be operated in accordance with Technical Specifications (TS) Sections 3.4.1.4, 3/4.7, 3/4.11 and 6.0. The iron and copper/lead steps will be performed in Mode 5 (Cold Shutdown). Mode 5 is defined as K_{eff} less than .99 and RCS temperature less than or equal to 200°F. During the SGCC iron step the RCS temperature will be maintained between approximately 190°F and 195°F. During the SGCC drying step the RCS temperature will be maintained at approximately 150°F. During the SGCC copper/lead step the RCS temperature will be maintained between approximately 110°F and 140°F. The RCS conditions for the SGCC iron step, drying step and copper / lead step are thus bounded by Mode 5 requirements. Some of the Technical Specifications discussed below, such as 3.4.4, 3.6.6.1 and 3.7.7 are applicable only in Modes 1, 2, 3 and 4 and so do not apply during the Mode 5 SGCC. They are addressed herein because there is the potential that the SGCC could result in changes that would impact the plant when power operation is resumed during Cycle 13.

Technical Specification 3.4.1.4 requires that at least two of the loops(s)/trains listed below shall be Operable and at least one reactor coolant and/or shutdown cooling loop shall be in operation in Mode 5:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump.
2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump.

3. Shutdown Cooling Train A.
4. Shutdown Cooling Train B.

Technical Specification Surveillance Requirement 4.4.1.4.2 requires that the required steam generator shall be determined Operable by verifying the secondary side water level to be ≥ 50 % of wide range indication at least once every twelve hours. For the SGCC iron step, drying step and copper/lead step, the steam generators are not operable because the water levels will be below 50 %. This is acceptable because both trains of shutdown cooling will be Operable and at least one train will be in operation.

UFSAR Section 9.3.6.3.4 discusses a Loss of Shutdown Cooling with the Reactor Coolant System Partially Filled. The UFSAR section was in response to Nuclear Regulatory Commission Generic Letter 87-12 dated July 9, 1987. In the event that shutdown cooling is lost during the time that the steam generators are inoperable, during SGCC, their operability can be restored by terminating the chemical cleaning and refilling the steam generators with auxiliary feedwater.

Technical Specification 3.7.2 requires that the temperature of the secondary coolant in the steam generators shall be greater than 115°F when the pressure of the secondary coolant is greater than 210 psig. The copper/lead step will be performed at approximately 110°F at a maximum steam generator secondary side pressure of 70 psig (Reference STI-W3-2003-0005-00). These parameters are within the requirements of the TS 3.7.2.

Technical Specification 3.4.4 requires that each steam generator shall be Operable in Modes 1, 2, 3 and 4. Operability is determined during shutdown by selecting and inspecting the steam generator tubes per Technical Specification Table 4.4-2. The purpose of the SGCC is to facilitate the steam generator tube inspection by removing the copper that may potentially interfere with the (eddy current) inspection. Following the proposed steam generator chemical cleaning, the steam generator tubes will be inspected per the Technical Specification requirements.

Technical Specification 3.11.1.4 sets limits on the quantity of radioactive material that will be stored in outside temporary tanks. The temporary storage and disposal of radioactive liquid waste water generated during chemical cleaning is evaluated in one of the separate 10 CFR 50.59 evaluations mentioned above (see 10CFR50.59 "Environmental" Exemption). The conclusion of the "Environmental" Exemption is that the liquid waste generated by the SGCC will be in compliance with this Technical Specification.

It should be noted, as previously stated, that the amount of toxic gases or fumes to be released should not present a personnel safety risk. Small quantities of ammonia and hydrazine will be released, but not in sufficient quantities to present a hazard. Calculations have been performed to determine the concentration and dispersion of the fumes. The calculation results show that the fume concentration is within all of the acceptable

regulatory and toxicity limits. See the separate 10 CFR 50.59 "Environmental" Exemption for a complete discussion on this subject.

Monitoring will also be conducted to determine the concentration of chemical fumes at key locations in and around Waterford 3 (see the separate 10 CFR 50.59 "Environmental" Exemption for a thorough discussion of this issue). Both fixed and mobile chemical detectors may be utilized for the monitoring. This monitoring will help to protect plant personnel from any adverse effects due to chemical release.

Information obtained from Nuclear Regulatory Commission (NRC) Information Notice (IN) 95-41, "Degradation of Ventilation System Charcoal Resulting From Chemical Cleaning Of Steam Generators" will be utilized during the Waterford 3 SGCC project to maintain compliance with the Waterford 3 Technical Specifications. The pertinent facts from this information notice are as follows:

1. In June 1994 the Surry Unit 2 Steam Generators were chemically cleaned using basically the same chemicals that will be used to clean the Waterford 3 steam generators (EDTA, EDA and hydrazine etc.).
2. Following the SGCC the steam generators were opened to allow access for sludge lancing. Concentrations of ammonia and hydrazine, in containment, were then measured at 30 ppm and 6 ppm respectively.
3. Both train A and train B of the auxiliary ventilation system were run, for approximately eight hours to reduce the chemical concentrations in containment.
4. The charcoal in both trains was then sampled via laboratory tests for the methyl iodide removal efficiency. The removal efficiency for trains A and B was 93.4 % and 90.7 % respectively. These efficiencies were below the Technical Specification requirements.

The conclusion in the Information Notice is that discharging the air involved in steam generator chemical cleaning operations through systems containing charcoal was likely to degrade the charcoal. The degradation mechanism was thought, by the NRC, to be breakdown of the hydrazine into water (along with EDA and EDTA as contributors). The water then degraded the methyl iodide removal efficiency of the charcoal.

During SGCC the Control Room Ventilation system will be placed in the "Isolate" mode. This will ensure the operability of the system by isolating the charcoal adsorber in the system from contamination and degradation due to contact with chemicals released during the SGCC. Technical Specification 3/4.7.6.1 is not applicable because this TS is applicable only in Modes 1, 2, 3 and 4. Note that there will be times during the SGCC when no chemical fumes will be released. During these times operation of the system may be resumed at the discretion of the Operations Department. This will enable the carbon dioxide levels in the control room to be maintained at acceptable levels.

Shield Building Ventilation (SBV) is to remain secured in Mode 5 when the Shield Building Hatch is closed. When in Mode 5 with the hatch open the Process Lead must verify that ammonia and hydrazine vapors are less than

one part per million (PPM) at the hatch opening before starting SBV. If the vapors are equal to or exceed 1 PPM then SBV start will be delayed until System Engineering recommends a course of action. If SBV is in operation and vapors are equal to or exceed 1 PPM then SBV will be secured and System Engineering will be notified to evaluate the need to sample the filter charcoal. This will ensure compliance with Technical Specification 3/4.6.6.1 even though this TS is only applicable in Modes 1, 2, 3 and 4. If a radiological release occurs the system will be allowed to run normally, whether or not there are chemical fumes present. This is shown to be acceptable per the following discussion:

The Loss of Coolant Accident (LOCA) and Non-LOCA scenarios must be considered with respect to charcoal filter degradation affects on event consequences. For a lower mode LOCA (Mode 4 or Mode 5) Combustion Engineering provided an Infobulletin on Emergency Core Cooling System (ECCS) requirements needed to mitigate the consequences of a LOCA (ABBCE Infobulletin No. 99-01). This Infobulletin stated that with one High Pressure Safety Injection (HPSI) train available within 10 minutes, a severance of the largest line connected to the RCS will not uncover the core or increase the fuel cladding temperature. This means that the typical design basis LOCA radiological source terms will not be present and the safety charcoal filters will not be needed to meet the typical LOCA design basis requirements. Thus the issue of potential charcoal filter degradation is not predicted to produce adverse consequences for the Mode 4 and 5 LOCA due to no predicted fuel failure.

For the lower mode Non-LOCA events, the expected RCS and SG activities (RCS < 0.01 uCi/gm as of 9/27/00) are a factor of 100 below those used in the accident analyses. Non-LOCA events initiated from lower mode operation are not predicted to incur fuel failure. Thus it is expected that potential charcoal filter degradation would have no affect on the accident consequences.

During SGCC the Controlled Ventilation Area System Filter Trains will be secured. Technical Specification 3/4.7.7 is only applicable in Modes 1, 2, 3 and 4. If the CVAS Filter Train must be placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV system. Also as described above the system will be placed in service in the event of a radiological release regardless of chemical vapors. Chemical monitoring will be conducted for chemicals at the outside air intake for the RAB Normal Supply System and the Reactor Auxiliary Building (RAB) – 4 wing area. Sampling of the charcoal in the Controlled Ventilation Area System (CVAS) will also be conducted after the completion of the SGCC if monitoring results indicate that sampling is required. Also as stated above chemical fume degradation of the charcoal would have no affect on the accident consequences for a Mode 4 or Mode 5 accident (SGCC is performed in Mode 5).

During SGCC the Fuel Handling Building (FHB) Emergency Ventilation will be secured. If the FHB Emergency Ventilation must be placed in service R1

chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV system.

For all of the above mentioned safety related, charcoal containing, HVAC systems the monitoring during SGCC in conjunction with potential (if chemical fumes are detected) charcoal sampling will ensure, and demonstrate compliance with the applicable Technical Specifications when full power operation is resumed following RF 12 (the SGCC will be performed in Mode 5 when the operation of the HVAC systems is not required by Technical Specification. If charcoal testing is warranted, due to fumes released during SGCC, and the testing reveals that any of the safety related charcoal adsorber beds is below the Technical Specification efficiency, the charcoal will be replaced prior to resuming full power operation. This ensures compliance with the applicable Technical Specifications after SGCC.

It should be noted that dedicated communication lines (telephones and/or radios) will be established linking the EOI Waterford Maneuvering Representative (definition from FANP Procedure Dwg. 6026845A) with the Control Room. The EOI Maneuvering Representative will be responsible for coordination of all activities. In addition, the Maneuvering Representative will be the single Waterford 3 point of contact for the evolution. This helps to minimize any risks from the chemical cleaning activity by providing immediate notification to the Waterford 3 Control Room of any potential problem.

The UFSAR accident analyses clearly bound the chemical cleaning evolution. The plant conditions that will be maintained during this evolution are RCS temperature maintained at a maximum of approximately 197°F, RCS pressure in accordance with the RCP operating curves and steam generator pressure and temperature at saturation conditions. Note that the saturation pressure corresponding to a temperature of 200 °F (upper limit of Mode 5) is approximately 11.5 psia. The stable plant conditions that are maintained during this process make it highly unlikely that a plant event would occur as a result of the chemical cleaning process. Comparison of these plant conditions to the conditions assumed in the UFSAR analyses demonstrates that the UFSAR analyses represent worst case scenarios and clearly bound the steam generator chemical cleaning evolution. No aspect of the chemical cleaning activity increases the probability of an accident that was previously analyzed in the UFSAR.

The Blowdown System, Condensate Makeup System, and Sump Pump System are not postulated to initiate any Chapter 15 accidents. Chapter 15 initiating events that have been identified as having the potential to be affected by the SGCC are as follows:

1. Increased Main Steam Flow (UFSAR Section 15.1.1.3)
2. Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (UFSAR Sections 15.1.1.4 and 15.1.2.4)
3. Steam System Piping Failures (UFSAR Sections 15.1.3.1, 15.1.3.2, 15.1.3.3)
4. Loss of Normal AC Power (UFSAR Section 15.2.1.4)
5. Primary Sample or Instrument Line Break (UFSAR Section 15.6.3.1)

6. Steam Generator Tube Rupture (UFSAR Section 15.6.3.2)
7. Liquid Waste System Leak or Failure (Release to Atmosphere – UFSAR Section 15.7.3.2)
8. Postulated Radioactive Releases Due to Liquid Containing Tank Failure (UFSAR 15.7.3.3)

Items 1 and 2: Increased main steam flow (item No. 1 above) can be caused by the inadvertent opening of an atmospheric dump valve (or safety valve). The failure scenario would be that the release of the chemical cleaning vapors through the ADVs during SGCC would damage the ADV (or safety valve) to such an extent that the ADV would fail to seat properly either during the SGCC or following RF 12 (during cycle 13 or following cycles). The vapors produced during the iron step will be composed primarily of water and ammonia with a small amount of hydrazine and other chemical cleaning chemicals (carryover from the steam generator) also present. The copper step vapors will be essentially all water and ammonia entrained in air with small amounts of the chemical cleaning chemicals (carryover from the steam generator). Materials evaluations, presented in FANP Document No. 51-5030944, show that this is not a credible failure mechanism. Similar justification can be used to rule out the inadvertent opening, due to component failure, of a Steam Generator Atmospheric Dump Valve (item No 2 above). As added justification of the acceptability of the SGCC process it should also be noted that there have been no SGCC induced failures of valves or piping at other plants that have cleaned their steam generators. Previous inspections, at other plants, have also not identified any unexpected (expected amounts of metal loss range from negligible for metal exposed to steam and gaseous cleaning chemicals to small amounts, within the design allowances, for metal exposed to the chemical cleaning liquids) material degradation following SGCC.

Item 3: From a purely engineering standpoint a steam System Piping failure is not very likely to occur during the Mode 5 SGCC because of the low temperatures and pressures associated with the SGCC. These piping breaks are postulated as part of the plant licensing basis in other modes (i.e. UFSAR Section 15.1.3.2 postulates steam system piping failures inside and outside containment in Mode 3 and Mode 4) and during the next operating cycle pressures and temperatures will be at higher operating levels i.e. nominal pressures and temperatures.

The SGCC iron steps, drying step and copper/lead steps will occur during Mode 5. The Technical Specifications define the Mode 5 average RCS coolant temperature as less than or equal to 200 °F. As stated above the saturation pressure at a temperature of 200 °F is approximately 11.5 psia. The maximum pressure in the steam lines during SGCC will therefore not exceed the pressure imparted from the FANP process (solution injection and draining and sparging). The upper limit on the SGCC process pressure is 70 psig (Reference STI-W3-2003-0005-00). The design pressure and temperature for the main steam lines is 1085 psig and 555 °F respectively. The maximum operating pressure and temperature is 985 psig and 545 °F

respectively. Immediately prior to the SGCC the main steam system will have been subjected to operating pressures and temperatures below the design conditions but much above the conditions that will be present during the cleaning. It is therefore extremely unlikely that the steam lines would fail at the SGCC pressure and temperature of 70 psig and 200 °F respectively.

The probability of a Steam System Piping Failure is not increased due to SGCC because the chemicals (in both liquid and vapor form) introduced into the piping during the SGCC will not reduce the strength of the piping. As stated above the main steam lines will have mostly water (as steam), ammonia and slight amounts of hydrazine and other chemical cleaning chemicals in them during SGCC. Materials evaluations, presented in FANP Document No. 51-5030944, show that there will be no detrimental effect on the steam system piping or valves.

Although a break in the steam generator blowdown piping is not an accident previously evaluated in the UFSAR, to completely evaluate the issue of potential pipe breaks this issue will also be addressed in the response to this Question. Additional FANP documents show that there will be no detrimental effect, due to corrosion and or erosion, on the Waterford 3 blowdown piping due to SGCC. The titles and identification numbers of these documents are as follows:

1. "Waterford Unit 3 Material and Geometry Review and Evaluation" (FANP Document Identifier 5008453-00).
2. "PVNGS – 2 Chemical Cleaning Materials Evaluation / Corrosion Estimates" (FANP Document Identifier 51-1228499-00, referenced in FANP Document 5008453-00).
3. "PV-1 HTCC Material Evaluation" (FANP Document Identifier 51-1234952-00, referenced in FANP Document 5008453-00).
4. "Salem Unit 2 Materials Evaluation" (FANP Document Identifier 51-1269270-00, referenced in FANP Document 5008453-00).
5. "300 F Chemical Cleaning Velocity Test Report" (FANP Document Identifier 51-1234107-00, referenced in FANP Document 51-1234952-00).
6. "PVNGS-1 High Temperature Chemical Cleaning Research Qualification Test" (FANP Document Identifier 51-1234937-00, referenced in FANP Document 51-1269270-01).
7. "PV-1 High Temperature Chemical Cleaning Optimization Test Report" (FANP Document Identifier 51-1234949-00, referenced in FANP Document 51-1269270-01).
8. Dominion Engineering, Inc. Report R-4135-00-1, Rev. 0, "Summary of Nominal Process and Low Deposit Loading Autoclave Tests for Qualification of the Siemens Chemical Cleaning Process at Waterford 3", August 2000.
9. "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report", (FANP Document Identifier 51-5030944-00), September 29,2003.
10. Dominion Engineering,Inc. Letter L-4160-01-2, "Material and Geometry Review – Waterford 3 Steam Generator Chemical Cleaning", September 15,2003.

The reports demonstrate that the corrosion/erosion of the SA-106 Grade B blowdown piping at Waterford 3, during SGCC, will be well within the allowable design limits. There is therefore no potential for the proposed SGCC to cause a break of the blowdown piping or blowdown nozzle.

The cleaning chemicals that will be injected into the steam generators through the blowdown piping will be in liquid form at approximately ambient atmospheric temperature (less than 100 °F). It should be noted that at temperatures below approximately 250 °F the metal/chemical reaction rate of the chemicals used in the cleaning process is very low. This low reaction rate, at low temperature, minimizes the impact that the chemicals have on the blowdown piping as they are injected into the steam generators.

The temporary FANP supplied lines, hoses and pumps do not create the possibility of a new High Energy Line break. The pressure and or temperature operating conditions of the temporary lines and components are below the HEL break threshold.

Item 4: Any SGCC affect on the probability of a Loss of Normal AC Power accident (item No. 4 above) can also be eliminated. The power supply for the vendor supplied pumps and major equipment is taken from outside of the Waterford 3 plant (an independent transformer is connected to the Entergy electrical grid outside of the plant protected area with power cables run into the protected area through the security zone). Internal plant electrical power is supplied only to limited vendor portable equipment from existing 110 volt receptacles. The 110-volt receptacles are not capable of initiating any UFSAR Chapter 15 accidents.

Item 5: Effects on the probability of a Primary Sample or Instrument Line Break (item No. 5 above) can also be eliminated. The stainless steel tubing and piping that is used on the primary system is unaffected by the chemicals used in the proposed SGCC process. The chemicals would therefore be incapable of causing a break even if they were to enter the primary system. It should be noted here that ER-W3-2003-0366-02 provides guidance for isolating some of the permanent plant instrumentation that has the potential to be affected by SGCC. This instrumentation is all on the secondary system. The isolated instrumentation will therefore not be impacted by the SGCC. For the permanent plant instrumentation that has the potential to be impacted but that is not isolated during SGCC, the ER also justifies that there will be no harm done to this instrumentation. See the ER for further information on this subject.

Item 6: The separate "Process" 10 CFR 50.59 Report addresses the effects of the SGCC on the steam generator tubes and steam generator internals (item No. 6 above).

Items 7 and 8: The separate "Environmental" 10 CFR 50.59 Report also evaluates the effect of SGCC on the potential liquid releases (items No. 7 and 8 above).

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 FSAR? ☐ Yes ☒ No

BASIS:

The probability of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. As stated above, the impact of the chemicals used in the SGCC on the Inconel 600 tubes that provide the barrier for fission product release will be addressed in a separate 10CFR 50.59 Report. There will be however no adverse impact on the steam generator tubes. Corrosion of the steam generators and affected plant systems materials will be well within the design corrosion allowances. Foreign material control will be aggressively pursued in accordance with existing plant administrative control procedures. All of the FANP supplied equipment, hoses etc. will be flushed and tested prior to use. The temporary equipment will be operated and the discharges will be sampled (in some cases with the use of filter bags) until the equipment is proven to be clear of any foreign material (dust, sand, scale etc). This ensures that no foreign material or loose parts, which could conceivably cause a tube failure during plant operation, would be inadvertently injected into the steam generator or other plant systems. The proposed steam generator chemical cleaning will not require that any plant structures, systems or components operate outside of their design bases.

The effect of the release of evaporated cleaning chemicals through the ADVs has been evaluated in FANP Document 51-5030944, "Waterford Unit 3 2003 *Chemical Cleaning Qualification Final Report*". That evaluation confirms that the constituents of the vapor that will be released through the ADV will have no adverse impact on the ADV materials. Therefore, failure of the ADV, as a result of SGCC, is not a credible event.

Temporary system connections to permanent plant equipment are easily isolated or disconnected in the event of an emergency. No temporary system connections are made within the Containment building. The location of all temporary system connections and the required chemical cleaning equipment have been chosen to minimize the potential impact on equipment important to safety.

The SGCC process/injection lines/hoses will carry the chemical cleaning chemicals, ammonia and hydrogen peroxide. These chemicals could have the potential to increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the UFSAR if the new chemicals could cause unanalyzed damage to safety related components that they may come in contact with during a potential process line/hose break. The following addresses this issue:

Effect of Chemical Cleaning Operations and Potential Impact of Off-Normal Conditions on Integrity and Operability of Safety Related Structures, Systems and Components at W3

The purpose of this evaluation is to assess the impact of RF12 steam

generator chemical cleaning operations on safety related equipment in the event that this equipment is exposed to liquid or gaseous species during normal or off-normal cleaning operations. Normal operations would include emissions associated with sparging air or nitrogen through the steam generators, and filling and draining of the chemical cleaning system including hoses, lines and tanks. Off-normal conditions would range from unanticipated fugitive emissions to liquid or gaseous exposures due to leaks or spills.

Prior to the chemical cleaning in RF10 (2000), a similar evaluation of the *environmental qualification (EQ) of safety related equipment* was conducted. In this evaluation, it was determined that normal and off-normal emissions would not represent a risk to EQ equipment. This was due to the fact that: (1) the two solvents used in RF10 (iron and copper solvent) had pHs that were within the bounds of that previously determined to be acceptable per the EQ program (4.5 to 10), and (2) air emissions from SGCC operations were too low to represent a corrosion risk to equipment.

For the chemical cleaning in RF12, iron and copper solvents will again be used. The iron solvent is still EDTA-based but applied at lower temperature, 190 to 195°F as opposed to 320 to 345°F, and without "venting" through the ADVs. Nitrogen sparging will be used instead of venting to promote mixing in the steam generators. The pH of the iron solvent will be 6.9 to 7.8 during preparation, and will be limited to 6.9 to 9.0 during application as described in FANP Document 6027358, "Waterford 3 2003 Chemical Cleaning Process Control Procedure". Accordingly, the pH is within the bounds of the EQ program (less than 10.5 but greater than 4.5). The emissions of hydrazine, and ammonia associated with sparging in lieu of venting are similar to those associated with the RF10 cleaning as documented in FANP Document 51-5030944 "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report" and FANP Document 51-5031596-01 "Waterford Unit 3 2003 Chemical Cleaning Emissions Estimates". As such, the SGCC iron step application is again considered acceptable with regard to exposure of safety related equipment to SGCC emissions or spills and leaks.

The copper solvent that will be used in RF12 is also similar to that used in RF10, a mixture of EDA and ammonium carbonate, with a small amount of EDTA. The only significant change to the solvent is that 20 g/l hydrogen peroxide (about 2 weight percent) will be added early in the process to improve copper dissolution kinetics. Emissions estimates performed by FANP show that the release of the only species with significant volatility, other than ammonia which was considered in RF10, in the copper step (EDA) due to air sparging is small and similar to that evaluated in RF10. The volatility of hydrogen peroxide is 300 times lower than that of EDA or ammonia, and its expected concentration in air either at the ADVs or adjacent to spills or leaks is very low (calculated to be only 3 ppm). At this concentration, peroxide is essentially non corrosive to ferritic and austenitic steels, and will not corrode typical materials of construction used

in instrumentation and controls (elastomers, plastics, rubber, Viton, etc.). As such, the SGCC copper step application is again considered acceptable with regard to exposure of safety related equipment to normal SGCC emissions.

Auxiliary Component Cooling Water (ACCW) Pump A is immediately adjacent to the inside of the wall penetration (elevation -27) through which the 4" FANP process line enters the Reactor Building. If the process line were to break and spray chemicals directly onto the pump motor, and if the chemicals were to enter the motor and contact the copper components in the motor, the motor would be severely damaged or destroyed. To protect the motor a scaffold barrier will be erected over the ACCW pump to prevent chemicals from entering the motor. The scaffold barrier will be erected in accordance with procedure PMC-002-006, "Erecting Scaffolding" and Special Test Instruction, STI-W3-2003-0005-00, "Special Test Instruction Waterford 3 Steam Generator Chemical Cleaning". The materials used to erect the scaffold barrier will either be chosen for their resistance to the SGCC chemicals or protected from degradation by the SGCC chemicals. The scaffold barrier will be constructed so as not to obstruct cooling air flow to the pump/motor.

In addition, while the nominal pH of the copper solvent is 10 as it was in RF10, FANP Document 6027358 "Waterford 3 2003 Chemical Cleaning Process Control Procedure" sets the allowable range at 9.5 to 12. This pH range in excess of 10.5 is outside the acceptable range of the EQ program of 4.5 to 10. The pH in excess of 10 is due to use of higher concentrations of EDA in the solvent to maximize solvent capacity for copper dissolution. Accordingly, a review of the corrosiveness of this additional EDA on typical safety related materials of construction was completed in preparation for the chemical cleaning in RF12. Corrosion rates for nine different materials due to exposure to EDA were obtained (silicon rubber, epoxies, polyolefins, natural rubber, carbon and low alloy steel, stainless steel, galvanized steel, heat shrink tubing, EPR and Viton). Corrosion rates for all non steel materials were essentially zero. Very slight corrosion to carbon and stainless steel was reported in the literature (less than 0.01 mils per hour). This is considered negligible, and as such exposing safety related equipment to a copper solvent as a result of a spill is considered acceptable. If a spill does occur a Condition Report will be initiated and the impacted equipment will be inspected and evaluated.

The only safety related portion of Blowdown is between BD-103A(B), (Blowdown Outside Containment Isolation Valve) and the steam generators. The proposed change does not affect the safety related portion of the blowdown system. The cleaning process design will limit corrosion of steam generator secondary side components and BD base material to within corrosion allowances. The proposed Special Test Instruction directs draining the steam generators using blowdown to Circulating Water (CW) or the Condenser Hotwell. In the case of the condenser hotwell, the BD system will be operated in accordance with normal operating procedures. These portions of the Blowdown system are not safety related.

The proposed Special Test Instruction provides instructions for sparging the steam generators with compressed air. The normal operating pressure of the steam generators bounds the 70 psig applied during sparging. Thus, the probability of a malfunction of the steam generators is unaffected by the proposed air sparging.

Nuclear Regulatory Commission (NRC) Information Notice (IN) 95-41 describes an event in which cleaning chemicals damaged Heating, Ventilation and Air Conditioning (HVAC) Systems equipped with charcoal filters. Since the Special Test Instruction may introduce cleaning chemicals into areas ventilated by systems containing charcoal filters, the probability of a malfunction of HVAC systems equipped with charcoal filters is potentially affected.

In order to protect the safety related charcoal in the plant Heating, Ventilation and Air Conditioning systems the following actions will be taken during the proposed Waterford 3 SGCC:

1. The Control Room Ventilation system will be placed in the "Isolate" mode.
2. Shield Building Ventilation (SBV) is to remain secured in Mode 5 when the Shield Building Hatch is closed. When in Mode 5 with the hatch open the Process Lead must verify that ammonia and hydrazine vapors are less than one part per million (PPM) at the hatch opening before starting SBV. If the vapors are equal to or exceed 1 PPM then SBV start will be delayed until System Engineering recommends a course of action. If SBV is in operation and vapors are equal to or exceed 1 PPM then SBV will be secured and System Engineering will be notified to evaluate the need to sample the filter charcoal.
3. The Controlled Ventilation Area System (CVAS) will be secured. If CVAS must be placed in service then the Process Lead must verify that the ammonia and hydrazine vapors are less than 1 PPM prior to starting CVAS. If the vapors are equal to or exceed 1 PPM then CVAS start will be delayed until System Engineering recommends a course of action. If CVAS is in operation and vapors are equal to or exceed 1 PPM then CVAS will be secured and System Engineering will be notified to evaluate the need to sample the filter charcoal.
4. During SGCC the Fuel Handling Building (FHB) Emergency Ventilation will be secured. If the FHB Emergency Ventilation must be placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV and CVAS systems. R1

Note that there will be times during the SGCC when there will be no likelihood of chemical fumes being released. During these times the operation of the various ventilation systems may be restored at the discretion of the Operations Department. It should also be noted, as previously stated, that the amount of toxic gases or fumes to be released should not present a personnel safety risk. Small quantities of ammonia, hydrazine and other cleaning chemicals (EDTA and EDA) will be released, but not in sufficient quantities to present a hazard. Monitoring will be conducted to determine the

concentration of chemical fumes at key locations. Both fixed and mobile monitors may be utilized. This monitoring will help to protect plant personnel from any adverse effects due to chemical release.

An increase in the probability of a Primary Instrument Line Break can also be eliminated. The stainless steel tubing and piping that is used on the primary system is unaffected by the chemicals used in the proposed SGCC process. The chemicals would therefore be incapable of causing a break even if they were to enter the primary system. It should be noted here that ER-W3-2003-0366-02 provides guidance for isolating some of the permanent plant instrumentation that has the potential to be affected by SGCC. This instrumentation is all on the secondary system. The isolated instrumentation will therefore not be impacted by the SGCC. For the permanent plant instrumentation that has the potential to be impacted but that is not isolated during SGCC, the ER also justifies that there will be no harm done to this instrumentation. See the ER for further information on this subject.

Vibration analyses have also demonstrated that there will be no deleterious effect on the steam generators from the air sparging during the copper/lead step (Reference Dominion Engineering calculation C-4133-00-2).

Based on the above discussion, the probability of a malfunction of a component important to safety is not increased.

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

BASIS:

The consequences of an accident previously evaluated in the UFSAR will not be increased. As described above the SGCC will be done in Mode 5 prior to the start of RF 12. During this time the RCS pressures and temperatures will be well below those assumed in the UFSAR Chapter 15 accident analyses. The relatively low energy of the plant systems, when compared to full power operations, ensures that the consequences of any event that may occur are well bounded by the existing analyses. The consequences of these previously evaluated accidents would thus be much less severe if they were to occur during the steam generator chemical cleaning. Also, the assumptions used for the loss of shutdown cooling events bound the plant conditions that will be established for chemical cleaning. In addition, this process will not introduce or significantly increase any source terms that are associated with the UFSAR analyses. The stable plant conditions that are required during the chemical cleaning process also reduce the risk of any credible accident. Therefore, the consequences of any analyzed accident are not increased.

As stated in the answer to Question No. 1 above the break of a main steam line is evaluated in the UFSAR. The consequences of the UFSAR evaluated break were based upon the steam lines carrying only steam and specific chemicals normally added to the secondary system. Also as explained above, during the SGCC process the main steam line will also carry some small amounts of the chemical cleaning chemicals, ammonia and hydrazine. These

chemicals could have the potential to increase the consequences of the previously evaluated accident if the new chemicals could cause unanalyzed damage to safety related components that they may come in contact with during the postulated pipe break. Breaks in the FANP piping/hoses used for the SGCC also have the potential to introduce the chemical cleaning chemicals into the plant. This issue was discussed in the answer to Question No. 2 above. The conclusion is that there will be no impact on plant safety related components.

In addition, as discussed in the answer to Question No. 1 above, the chemicals used in the SGCC process are not very reactive at low temperatures. Therefore even if a leak were to occur the chemicals would soon cool to the ambient temperature (temperatures inside containment are expected to be higher than temperatures in the RAB but both will be sufficiently low to lower the chemical reactivity) and this would limit any impact on plant systems, structures or components. Systems, structures or components that could potentially be exposed to the chemicals, due to a piping rupture, will therefore suffer no consequences that are not bounded by the existing analyses in the UFSAR.

Also as discussed in the answer to Question No. 2 above Auxiliary Component Cooling Water (ACCW) Pump A is immediately adjacent to the inside of the wall penetration (elevation -27) through which the 4" FANP process line enters the Reactor Building. If the process line were to break and spray chemicals directly onto the pump motor, and if the chemicals were to enter the motor and contact the copper components in the motor, the motor would be severely damaged or destroyed. To protect the motor a scaffold barrier will be erected over the ACCW pump to prevent chemicals from entering the motor.

Upon return to full power operation, the consequence of an accident previously evaluated in the UFSAR is not increased since all changes to the secondary system, as a result of the SGCC process, will be within the design allowable limits. Any corrosion that will occur to the steam generators or affected plant systems will be well within design corrosion allowances. As stated above these effects are addressed in a separate 10 CFR 50.59 Report. The integrity of the affected plant systems will thus be maintained. Also, the removal of deposit from the steam generator tubes will help to ensure steam generator integrity by cleaning possible crack initiation and propagation sites. It should be noted however that the post SGCC eddy current test results will probably contain anomalies with reference to the previous eddy current results. This is because the SGCC will remove deposits from the tube surfaces that may currently be masking existing cracks in the tubes. Following the removal of the deposits these existing defects may become more readily detectable. This masking effect was first seen at the Millstone 2 plant in 1985. Nuclear Regulatory Commission Information Notice 85-37 contains information on this phenomenon. At Millstone essentially all of the masked defects were found within the region of the sludge pile (within thirteen inches of the tube sheet). This masking phenomenon should not be as extensive at Waterford 3 due to the smaller amounts of sludge in the

Waterford 3 steam generators relative to the amount at Millstone. Nevertheless it is expected that Waterford 3 could experience the masking phenomena during eddy current testing following the SGCC. As stated above steam generator eddy current testing will be performed in accordance with Technical Specification 3.4.4.

The chemical cleaning evolution has been developed to ensure effective chemical cleaning of the steam generators while minimizing corrosion attack. Laboratory qualification testing (as addressed in a separate 10 CFR 50.59 Report) has demonstrated that corrosion of steam generator internal materials due to the chemical solvent is minimal. FANP document No. 51-5008453-00, "Waterford Unit 3 Material and Geometry Review shows that the corrosion of the plant piping (main steam, blowdown), valves (ADV, MSIV) and components is also minimal. Corrosion monitoring measurements that will be taken following the SGCC will ensure that all affected materials remain within their corrosion allowance. Eddy current testing of the steam generator tubing and the inspection of key steam generator components such as the egg crates will ensure equipment and component integrity following SGCC. Therefore, the consequence of any analyzed accident is not increased.

As described above in the answer to Question No. 1 manual isolation of the control room ventilation system assures the operability of the system and the protection of the Operators from adverse effects due to potential chemical fumes. Therefore, the control room environment is not affected by the implementation of this process and the consequences of a toxic chemical accident are not increased.

The proposed Special Test Instruction also includes instructions for ensuring that the quantity of emergency breathing air for the Control Room is adequate, prior to start of the SGCC. It also includes instructions that the breathing air compressors are not to be run during SGCC. This will ensure that the emergency breathing air supply is available and is not contaminated during SGCC.

During SGCC the Control Room Ventilation system will be placed in the "Isolate" mode. This will ensure the operability of the system per Technical Specification 3/4.7.6.2 by isolating the charcoal adsorber in the system from contamination and degradation due to contact with chemicals released during the SGCC. Note that there will be times during the SGCC when no chemical fumes will be released. During these times operation of the system may be resumed at the discretion of the Operations Department. This will enable the carbon dioxide levels in the control room to be maintained at acceptable levels.

Shield Building Ventilation (SBV) is to remain secured in Mode 5 when the Shield Building Hatch is closed. When in Mode 5 with the hatch open the Process Lead must verify that ammonia and hydrazine vapors are less than one part per million (PPM) at the hatch opening before starting SBV. If the vapors are equal to or exceed 1 PPM then SBV start will be delayed until System Engineering recommends a course of action. If SBV is in operation and vapors are equal to or exceed 1 PPM then SBV will be secured and

System Engineering will be notified to evaluate the need to sample the filter charcoal. This will ensure compliance with Technical Specification 3/4.6.6.1 (Note: this TS is only applicable in Modes 1, 2, 3 and 4 and SGCC will be performed in Mode 5). If a radiological release occurs the system will be allowed to run normally, whether or not there are chemical fumes present. This is shown to be acceptable per the following discussion:

The Loss of Coolant Accident (LOCA) and Non-LOCA scenarios must be considered with respect to charcoal filter degradation affects on event consequences. For a lower mode LOCA (Mode 4 or Mode 5) Combustion Engineering provided an Infobulletin on Emergency Core Cooling System (ECCS) requirements needed to mitigate the consequences of a LOCA (ABBCE Infobulletin No. 99-01). This Infobulletin stated that with one High Pressure Safety Injection (HPSI) train available within 10 minutes, a severance of the largest line connected to the RCS will not uncover the core or increase the fuel cladding temperature. This means that the typical design basis LOCA radiological source terms will not be present and the safety charcoal filters will not be needed to meet the typical LOCA design basis requirements. Thus the issue of potential charcoal filter degradation is not predicted to produce adverse consequences for the Mode 4 and 5 LOCA due to no predicted fuel failure.

For the lower mode Non-LOCA events, the expected RCS and SG activities (RCS < 0.01 uCi/gm as of 9/27/00) are a factor of 100 below those used in the accident analyses. Non-LOCA events initiated from lower mode operation are not predicted to incur fuel failure. Thus it is expected that potential charcoal filter degradation would have no affect on the accident consequences.

During SGCC the Controlled Ventilation Area System Filter Trains will be secured. Technical Specification 3/4.7.7 is only applicable in Modes 1, 2, 3 and 4. If the CVAS Filter Train must be placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV system. Also as described above the system will be placed in service in the event of a radiological release regardless of chemical vapors. Chemical monitoring will be conducted for chemicals at the outside air intake for the RAB Normal Supply System and the Reactor Auxiliary Building (RAB) – 4 wing area. Sampling of the charcoal in the Controlled Ventilation Area System (CVAS) will also be conducted after the completion of the SGCC if monitoring results indicate that sampling is required. Also as stated above chemical fume degradation of the charcoal would have no affect on the accident consequences for a Mode 4 or Mode 5 accident (SGCC is performed in Mode 5).

During SGCC the Fuel Handling Building (FHB) Emergency Ventilation will be secured. If the FHB Emergency Ventilation must be placed in service chemical monitoring and System Engineering notifications/evaluations will be performed as described above for the SBV and CVAS systems. R1

For all of the above-mentioned safety related, charcoal containing, HVAC systems the monitoring during SGCC in conjunction with potential charcoal

sampling (if high enough levels of chemical fumes are detected) will ensure that there is no degradation during the proposed SGCC. As stated above and in the answer to Question No. 1 all applicable safety related charcoal adsorber Technical Specifications will also be maintained during and after the SGCC. The safety-related charcoal adsorber beds will therefore be capable of performing their safety function during and following the proposed SGCC. Hence the consequences of a radiological release accident will not be increased due to charcoal degradation due to the proposed SGCC.

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 FSAR? ☐ Yes ☒ No

BASIS:

The consequences of a malfunction of equipment important to safety will not be increased. Cleaning will take place during Mode 5. Both shutdown cooling trains will be operable and available for heat removal. Both steam generators will be completely drained in Mode 5 and hence will not be operable but the two shutdown cooling trains will be operable and available for heat removal. Since the steam generator chemical cleaning process will have minimal impact on the fission product barrier of the steam generator tubes and tubesheet (see separate 10 CFR 50.59 "Process" Exemption), the potential failure of this barrier and the radiation release consequences associated with that failure will not be changed by the proposed SGCC. As discussed above the plant conditions (pressure, temperature, core decay heat) during the steam generator chemical cleaning process will be much less than the conditions assumed in the UFSAR Chapter 15 accident analyses. Any release would thus be bounded by the previous analyses, ie the UFSAR analyses that evaluate loss of SDC events, loss of off-site power, station blackout, toxic gas release, steam generator tube rupture and failure of the steam generator secondary side, bound the consequences of any credible accident scenario associated with chemical cleaning. Laboratory testing has demonstrated that there will be no adverse impact on the steam generator tubes and that the amount of material removed from the other internal steam generator components will be within the design allowances (see separate 10 CFR 50.59 "Process" Exemption). A separate 10 CFR 50.59 "Environmental" Exemption will evaluate the SGCC impact on the potential radiation release due to discharging the spent chemicals (including the potentially radioactive dissolved steam generator deposits) to the FANP waste tanks.

All chemical cleaning equipment will be removed from the site and all involved plant systems will be returned to their original condition except that the steam generators will have less corrosion products and will have been exposed to the chemical cleaning solutions. Exposure to chemical cleaning solutions does result in minimal corrosion of the steam generators; however, corrosion will be monitored, and the corrosion increase due to chemical cleaning will not be allowed to exceed the design corrosion allowances. As stated above this is addressed in a separate 10 CFR 50.59 "Process" Exemption. Therefore, the probability of a malfunction or an increase in the consequences of a

malfunction of equipment important to safety is not increased.

The proposed procedure changes have no impact on the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR.

In Mode 5, the steam generators are not required as a heat sink if both Shutdown Cooling Trains are operable (per Technical Specification 3.4.1.4). The steam generators function as a heat sink for the RCS. However, while in Mode 5 during Steam Generator Chemical Cleaning, the SDC system will act as the RCS heat sink. Since the steam generators may be drained during portions of the SGCC, increased reliance is placed on the SDC system. The proposed change does not alter the SDC system or the operation of the SDC system. The proposed change does not alter the redundancy and reliability of the SDC system. Since SDC reliability is unaffected by the proposed change, the consequences of a malfunction of the steam generators are not affected by the proposed change. Additionally, as discussed in the answer to question No. 1 above, in the event that shutdown cooling is lost when the steam generators are inoperable the SGCC will be terminated and auxiliary feedwater will be used to restore steam generator operability. Also as described above steam generator structural integrity is unaffected by the proposed SGCC. Therefore the radiological consequences of accidents evaluated in the SAR are unaffected.

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

BASIS:

The possibility of an accident of a different type than any previously analyzed in the UFSAR is not created. The UFSAR Chapter 15 analysis bound any potential accident associated with chemical cleaning. Spills of the chemical are bounded by toxic release analysis (addressed in a separate 10 CFR 50.59 "Environmental" Exemption). The chemicals associated with chemical cleaning are organic in nature and are readily absorbed by charcoal filtration. In addition, the chemicals released by the SGCC are the same as the chemicals that can potentially be released offsite by the nearby chemical plants as identified in the UFSAR. There are thus no "new" chemicals associated with SGCC. As stated above (see the answer to Question No. 1) monitoring and sampling of the charcoal, if chemicals are detected by the monitoring, will ensure the operability of the safety related HVAC systems. Also as stated in the answer to Question No. 1 above the impact of the SGCC chemicals on the blowdown lines will be acceptable.

Radiological release of an outside tank's contents is bounded by the radiological analyses performed in Chapter 11, 12 and 15 of the UFSAR (addressed in a separate 10 CFR 50.59 "Environmental" Exemption). The UFSAR chapter 15 analyses bound all other credible events associated with chemical cleaning of the steam generators, including toxic gas release.

The qualification testing and systems evaluation that was performed during

the development of this process ensured that no new accidents or malfunctions that were outside of the UFSAR analyses were generated. In fact, the chemical cleaning process equipment can be quickly and easily isolated from the permanent plant equipment if necessary. This feature allows immediate mitigation of any event that may be initiated by the temporary chemical cleaning equipment. In addition, laboratory testing has shown that it is also acceptable for the chemical cleaning solution to remain in a steam generator for approximately twenty-four hours if necessary. The chemical reaction is self limiting in that only the amount of chemicals necessary to remove the maximum expected deposits are added to the steam generators. Once the chemicals react with the deposits they are used up and any further reaction is minimal. As discussed previously, this is further addressed in a separate 10 CFR 50.59 "Process" Exemption. Therefore, the possibility of an accident of a different type than any previously evaluated in the UFSAR is not created.

An accident of a different type than those evaluated in the UFSAR will not be created. The only change in the plant is the way the Unit is operated during the Mode 5 SGCC sequence. All previously postulated Chapter 15 events, whether they are increase in heat removal events, or decrease in heat removal events, etc., bound the events which could possibly occur during the SGCC. Most of the events presented in the UFSAR correspond to events that can occur with the Unit at 100% power. At lower modes, such as Mode 5, only steam line failure events could have significant consequences and as discussed above this type of failure is currently evaluated in the UFSAR.

As described above in Description of Proposed Change the proposed Waterford 3 Steam Generator Chemical Cleaning Project requires numerous new system connections between the temporary FANP supplied equipment and permanent Waterford 3 components. Also as described above the new connections have been authorized by previously approved Procedure PMC-004-008, "Installation Procedure Steam Generator Chemical Cleaning Equipment Installation". The new connections that have not been previously approved are the Blowdown Connections.

As stated above the blowdown connections are in the non safety portion of the blowdown system and can be easily isolated from the safety related portions of the blowdown system. These connections will be used to inject the chemicals into the blowdown piping (through piping/hoses running into the plant from the FANP tank farm) and then into the steam generator through the blowdown ring. These connections therefore have no impact on the safety related portions of the system.

ER-W3-2003-0366-02 provides an evaluation of the impact on the permanent plant instruments that are not isolated (are exposed to the cleaning chemicals) during SGCC. It also evaluates the impact of isolating various permanent plant instruments during SGCC. One of the instruments that is isolated and evaluated in the ER is the SG blowdown radiation monitor. See the ER for additional information.

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 FSAR? ☐ Yes ☒ No

BASIS

There are no other potential malfunctions based on SGCC process implementation. Equipment reviews, calculations, laboratory testing and justifications have been performed. The conclusion is that there is no compromise of the pressure boundary piping, valves, seals, or vessels installed as permanent plant equipment either during or after SGCC. ER-W3-2003-0522-00 evaluates the consequences of introducing SGCC chemicals into the primary system due to steam generator tube leakage between the secondary and primary systems (at times during the SGCC the pressure in the secondary system will be higher than the pressure in the primary system and there is existing steam generator tube leakage in SG No. 1). The conclusion of the ER is that the expected chemical concentrations in the primary system are too low to cause any problems with the RCS materials or fuel. All credible plant equipment malfunctions have been addressed throughout the SGCC 10 CFR 50.59 evaluations/reports (as discussed above the three applicable evaluations/reports are the "Implementation" Evaluation, the "Process" Exemption and the "Environmental" Exemption).

Failure modes of the operation of the temporary chemical cleaning equipment is addressed in a separate 10 CFR 50.59 Report (see the 10CFR50.59 Environmental Exemption).

As discussed in the answer to Question No. 1 above ER-W3-2003-0366-02 provides guidance for isolating some of the permanent plant instrumentation that has the potential to be affected by SGCC. The isolated instrumentation will therefore not be impacted by the SGCC. For the permanent plant instrumentation that has the potential to be impacted but that is not isolated during SGCC, the ER also justifies that there will be no harm done to this instrumentation. The ER also evaluates the effect of the steam generator vacuum drying on the permanent plant instrumentation.

The steam generator blowdown piping from the steam generators to the containment isolation valves is Nuclear Safety Class 2. Outside of the containment the blowdown piping is Non Nuclear Safety. As stated in the answer to Question No. 1 above a break in the steam generator blowdown piping in not evaluated in Chapter 15 of the UFSAR. The potential break of the blowdown piping is bounded by the break of a main steam line. The main steam line break is bounding because of the relative size differences (forty-two inches for the main steam line vs. two inches for the blowdown line) of the two lines. A potential break in the safety class 2 blowdown piping caused by SGCC would therefore be a potential malfunction of equipment important to safety of a different type than any previously evaluated in the UFSAR. As discussed in the answer to Question No. 1 above the following FANP documents show that the impact, from SGCC, on the blowdown piping will be acceptable:

1. "Waterford Unit 3 Material and Geometry Review and Evaluation" (FANP Document Identifier 5008453-00).
2. "PVNGS – 2 Chemical Cleaning Materials Evaluation / Corrosion Estimates" (FANP Document Identifier 51-1228499-00, referenced in FANP Document 5008453-00).
3. "PV-1 HTCC Material Evaluation" (FANP Document Identifier 51-1234952-00, referenced in FANP Document 5008453-00).
4. "Salem Unit 2 Materials Evaluation" (FANP Document Identifier 51-1269270-00, referenced in FANP Document 5008453-00).
5. "300 F Chemical Cleaning Velocity Test Report" (FANP Document Identifier 51-1234107-00, referenced in FANP Document 51-1234952-00).
6. "PVNGS-1 High Temperature Chemical Cleaning Research Qualification Test" (FANP Document Identifier 51-1234937-00, referenced in FANP Document 51-1269270-01).
7. "PV-1 High Temperature Chemical Cleaning Optimization Test Report" (FANP Document Identifier 51-1234949-00, referenced in FANP Document 51-1269270-01).
8. Dominion Engineering, Inc. Report R-4135-00-1, Rev. 0, "Summary of Nominal Process and Low Deposit Loading Autoclave Tests for Qualification of the Siemens Chemical Cleaning Process at Waterford 3", August 2000
9. "Waterford Unit 3 2003 Chemical Cleaning Qualification Final Report", (FANP Document Identifier 51-5030944-00), September 29, 2003.
10. Dominion Engineering,Inc. Letter L-4160-01-2, "Material and Geometry Review – Waterford 3 Steam Generator Chemical Cleaning", September 15,2003.

A break in the safety related blowdown piping (or the non safety related blowdown piping) can therefore not be created by the proposed Steam Generator Chemical Cleaning.

Also as discussed in the answer to Question No. 1, 2 and 3 above the effect of the cleaning chemicals (either liquids or vapors) has been evaluated and determined to cause no harm to any equipment, both inside and outside of containment, that is important to safety. There is therefore no effect, even from the UFSAR postulated mainsteam line break, on plant safety related structures, systems and components.

Vibration analyses have also demonstrated that there will be no deleterious effect on the steam generators from the nitrogen/air sparging during the SGCC (Reference Dominion Engineering calculation C-4133-00-2).

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

BASIS:

The steam generator tubes are a fission product barrier as described in the FSAR. The steam generator tubes are part of the reactor coolant system pressure boundary. The fuel cladding and the main steam isolation valves

(containment isolation valves) are also fission product barriers described in the FSAR.

Steam Generator Chemical Cleaning (SGCC) is an established foreign and domestic nuclear industry practice. The Electric Power Research Industry endorses chemical cleaning (ref EPRI TR-104553) and a body of Steam Generator chemical cleaning experience exists. The Waterford 3 SGCC process was laboratory tested in the FANP laboratory and confirmatory testing was performed in a separate laboratory by Dominion Engineering (see References above in the answer to Question No. 6). As discussed above the proposed SG cleaning chemicals will not adversely impact the S/G Inconel 600 tube material (see separate 10CFR50.59 Exemption Report), i.e. there will be no adverse corrosion of the tubes. Therefore the proposed chemical cleaning will not significantly decrease the wall thickness of the steam generator tubes and there will be no adverse impact on this fission product barrier.

Qualification testing and material evaluation studies (as described herein and in the "Process" 10 CFR 50.59 Exemption) have demonstrated that the base metal corrosion of the steam generator components and plant systems (including main steam isolation valves) that are exposed to chemical solvent, in both liquid and vapor form, is minimal and well within the design corrosion allowances. The proposed SGCC therefore has no impact on the steam generator components and plant systems that are exposed to the cleaning chemicals and decomposition products.

Following completion of the SGCC the steam generator tubes will be eddy current tested to ensure steam generator integrity. Any tube that has been determined to be defective will be removed from service. As stated in the answer to Question No. 2 above there has been a previous history, in the nuclear industry, of steam generator deposits masking defects in the steam generator tubes from detection during eddy current testing. It is expected that this same phenomena will occur at Waterford 3 during the eddy current testing that will be conducted following the proposed SGCC. These potential defects however, will not be caused by the Waterford 3 SGCC. The potential defects will have been pre existing and will merely be made detectable by the steam generator chemical cleaning process (steam generator tube deposit removal). The pre SGCC laboratory qualification testing, corrosion monitoring and post SGCC tube eddy current testing all serve to ensure compliance with Technical Specification 3/4.4.4. See the separate 10 CFR 50.59 "Process" Exemption for further information on the effects of the proposed SGCC on the steam generator internals. Therefore, structural integrity of the steam generator tubes is assured and the design basis limits for the steam generator tubes as defined in Technical Specification 3/4.4.4 are not reduced.

No temporary Steam Generator Chemical Cleaning equipment will be used inside containment.

The fuel cladding is also a fission product barrier as described in the FSAR. ER-W3-2003-0522-00 addresses the impact of SGCC chemicals on Fuel

and Reactor Coolant System Components due to leakage, through the steam generator tube(s), from the secondary to primary side during the chemical cleaning. The conclusion of the ER is that there will be no impact on either the fuel of reactor coolant system components.

Based on the information provided in this evaluation and the References, the chemical cleaning of the Waterford 3 steam generators does not pose any significant impact on plant nuclear safety, is well bounded by the safety analyses of the UFSAR and does not reduce the safety as defined in the basis of the Technical Specifications. The chemical cleaning process has been qualified and designed for implementation such that it will not result in safety issues and will have an overall minimal impact on the plant. The process will remove deposits from the steam generator tubes in such a way as to not degrade the structural integrity of the components or impact the health and safety of the public. Therefore the design basis limits for the fuel cladding, RCS pressure boundary and containment are not reduced by the proposed chemical cleaning.

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

BASIS:

The proposed Refuel 12 Steam Generator Chemical Cleaning is a maintenance activity to remove iron and copper/lead deposits in the Waterford 3 steam generators. If not removed the copper/lead deposits have the potential to interfere with the eddy current inspection of the steam generator tubes as required by Technical Specification 3/4.4.4. The proposed SGCC does not impact the design basis of the steam generators. No steam generator design basis calculations are impacted by the proposed change.

I. OVERVIEW / SIGNATURES

Facility: Waterford-3

Document Reviewed: ER-W3-2003-0135-000 **Change/Rev.** 0

System Designator(s)/Description: (INI) – Incore Nuclear Instrumentation System

Description of Proposed Change

Westinghouse Advisory letter NSAL-01-7 identified that In-Core Instrumentation (ICI) thimbles may experience growth which can create unanticipated loads on, and prevent re-assembly of, the instrumentation nozzle pressure boundary connection. To address this irradiation-induced growth of zircaloy incore instrumentation thimbles associated with the reactor vessel, a Westinghouse designed spacer will be installed under the seal carrier assembly and the upper Quickloc flange if growth is identified by field measurement. The spacer will be installed on all assemblies which indicate thimble growth could adversely affect assembly clearances. The spacer will eliminate the undesired "hard" connection between the upper internals and the reactor head instrumentation nozzles. A note will be added to FSAR Figure 3.9-18 to denote that a spacer may be installed on the in-core instrumentation nozzle to accommodate thimble growth.

In addition, since the addition of spacers will allow for additional ICI thimble growth, the ICIs themselves could move up within the core by an amount as great as the vertical size of the spacers. The ICIs include Core Exit Thermocouples (CETs), part of the Inadequate Core Cooling Instrumentation (ICCI) system, and Fixed Incore Detectors (FIDs). The Heated Junction Thermocouples (HJTCs), could also be affected by the addition of spacers, because the HJTCs are attached to the ICI support assemblies. The CETs and HJTCs provide information to the operators and FIDs are input to the Core Operating Limit Supervisory System (COLSS) and CE Core Operating Report (CECOR) program used for core monitoring. COLSS and CECOR will be modified to reflect any change in FID vertical position within the core: COLSS via addressable constants and CECOR via database changes. In the interim before the COLSS constant changes are made, a conservative penalty factor designed to capture the effect of different FID positions on the existing COLSS calculation will be inserted into COLSS to ensure that COLSS is able to perform its design function.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____.
(Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: 03-043)	Sections I, II, III, IV, and VI required

Preparer:

Howard Brodt / Howard L Brodt / EOI / S&EA 11/14/03

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

Reviewer:

Ed Wiegert / Edward D Wiegert / EOI / S&EA 11/14/03

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

OSRC:

K. Peters 11/14/03

Chairman's Name (print) / Signature / Date

[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
John Russo	Mechanical Portions
Thomas Hempel	Mechanical Portions
Rich Finch	Mechanical Portions

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	DRN 03-1921; FSAR Figure 3.9-18
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The Waterford-3 License research System (LRS) Entergy Fulfind electronic search was utilized for the License Basis Document applicability determination as detailed in the references in Section C below. The proposed change recommended by Westinghouse to add a spacer between the Incore Instrument (ICI) Assembly seal carrier and upper Quickloc flange will eliminate the interference between the upper internals and reactor head instrumentation nozzles. This modified arrangement will be annotated on FSAR Figure 3.9-18, In-core Instrumentation Nozzle.

The proposed spacer will not impact any component's ability to perform an intended design function. The Safety Evaluation Report, Technical Specifications and Technical Requirements Manual were reviewed and found unaffected. No special tests other than the normal post refuel initial service leak tests are required. The Waterford-3 facility currently has no provisions for dry fuel storage.

The potential increase in the vertical height of ICIs might possibly affect several systems or components described in the FSAR: CETs, HJTCs, COLSS, and CECOR, as described in Section I, Description of Proposed Change. As a result, these systems/components require further evaluation to determine the impact of the proposed change.

The document searches detailed in the following section describe the FSAR Sections and LDBs reviewed for impact by the proposed change.

C. **References**

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:

LBDS_50_59

Keywords:
"in-core detector", "in core detector", "ICI", "in-core instrument", "in core instrument", COLSS, CECOR, CET, HJTC

LBDs/Documents reviewed manually:

FSAR Section 3.9.5, Reactor Pressure Vessel Internals

FSAR Section 7.7.1.5, Core Operating Limit Supervisory System

FSAR Section 7.7.1.7, In-Core Instrumentation System

FSAR Figure 3.9.18, In-Core Instrument Nozzle

Technical Specification Section 3/4.3, Instrumentation

Technical Requirements Manual Section 3/4.3.3.2, Incore Detectors

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☒ Yes
☐ No

If "Yes," list the required changes.

(1) The CECOR code must be updated by Echelon Nuclear Engineering with actual ICI positions from any spacer additions before CECOR is used in Cycle 13 power ascension testing.

(2) Impacts on core power distribution parameters affected by the actual ICI positions (particularly ASI) must be addressed with interim COLSS penalty factors, and any other interim adjustments recommended by Westinghouse, must be added to COLSS to represent changed ICI positions before COLSS is declared operable. (These interim changes are intended to bound the possible effects of changed ICI positions until the revised COLSS setpoint analysis is completed.) The changes will need to be submitted in a form that addresses both the pre and post refuel startup test measurement values of these factors as well as startup test predictions.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) _____ ☐ No

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. **Executive Summary** (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

To address irradiation-induced growth of zircaloy incore instrumentation thimbles associated with the reactor vessel, a Westinghouse designed spacer will be installed under the seal carrier assembly and the upper Quickloc flange if growth is identified by field measurement. The spacer will be installed on all assemblies which indicate thimble growth could adversely affect assembly clearances. The spacer will eliminate the undesired "hard" connection between the upper internals and the reactor head instrumentation nozzles. A note will be added to FSAR Figure 3.9-18 to denote that a spacer may be installed on the in-core instrumentation nozzle to accommodate thimble growth.

In addition, since the addition of spacers will allow for additional ICI thimble growth, the ICIs themselves could move up within the core by an amount as great as the vertical size of the spacers. The ICIs include Core Exit Thermocouples (CETs), part of the Inadequate Core Cooling Instrumentation (ICCI) system, and Fixed Incore Detectors (FIDs). The Heated Junction Thermocouples (HJTCs), could also be affected by the addition of spacers, because the HJTCs are attached to the ICI support assemblies. The CETs and HJTCs provide information to the operators and FIDs are input to the Core Operating Limit Supervisory System (COLSS) and CE Core Operating Report (CECOR) program used for core monitoring. COLSS and CECOR will be modified to reflect any change in FID vertical position within the core: COLSS via addressable constants and CECOR via database changes. In the interim before the COLSS constant changes are made, a conservative penalty factor designed to capture the effect of different FID positions on the existing COLSS calculation will be inserted into COLSS to ensure that COLSS is able to perform its design function. The changes to CECOR will be made prior to start up testing so the confirmation of the Cycle Independent Shape Annealing Matrix, Radial Peaking Factor Measurements and Power Distribution Tests will be unaffected by this change.

Reason for proposed Change:

Westinghouse Advisory letter NSAL-01-7 identified that In-Core Instrumentation (ICI) thimbles may experience growth which can create unanticipated loads on, and prevent re-assembly of, the instrumentation nozzle pressure boundary connection.

50.59 Evaluation summary and conclusions

In summary, the addition of spacers to the ICI flange assemblies will not affect their design function, nor the design function of associated instrumentation (ICIs, CETs, HJTCs), COLSS, or CECOR. The change will not affect the probability or consequences of an accident described in the FSAR, nor introduce the possibility of a new type of accident. The change will not affect the probability or consequences of a malfunction, nor create the possibility of a malfunction with a different result. Finally, the change does not represent a departure from a method of evaluation described in the FSAR. Therefore, the proposed change does not involve an unreviewed safety question under 10CFR50.59.

B. License Amendment Determination

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 FSAR? ☐ Yes ☒ No

BASIS:

This change does not directly affect any structure, system, or component that is an accident initiator in the FSAR. These accident initiators are events such as changes in feedwater flow or temperature, changes in steam flow, secondary plant failures, SG tube ruptures, and RCS piping ruptures. The only effects of the change in ICI positions that could in any way be related to the accident initiators evaluated in the FSAR is the effects on COLSS or on the CPC azimuthal tilt calculation. As described in more detail for Question 2, the effect of ICI axial shift on the POLs and ASI in COLSS will be accomplished via addition of an adjustment to the COLSS EPOL and / or UNCERT terms, and COLSS/CPC tilt will be maintained conservative by use of the core monitoring program CECOR, which will be modified to accurately calculate core power distribution for the modified ICI positions. Westinghouse estimates that in the interim before the COLSS setpoint analysis is redone, a 4-5% penalty on thermal margin will result from the addition of a conservative EPOL adjustment. After the setpoint analysis is completed, the long-term adjustment on EPOL is expected by Westinghouse to be about a 1-2% impact on thermal margin. According to Westinghouse and Waterford Reactor Engineering, the expected thermal margin without the ICI spacer mod is in the range of 10-11%. Thus, even the larger interim EPOL penalty will preserve sufficient margin to safely operate the reactor at full power.

The purpose of the spacer is to maintain the original design function of the seal carrier assembly while accommodating ICI thimble growth. The instrument assemblies perform a passive safety related function by providing an RCS pressure boundary. This modified configuration will still meet the original design and ASME Section III Class 1 requirements. The design function of the seal carrier assembly will not be changed. Installation of the spacer will ensure that the original design function of the assembly will be met in the event of ICI thimble growth. Therefore, this change will have no impact on the design function of the assembly.

Since COLSS only provides assistance to the operators in maintaining the thermal margin-related tech spec LCOs, a reduction in margin is not a malfunction of COLSS (it still conservatively performs its design function). This reduction in thermal margin could potentially cause a minimal impact on the likelihood of an uncomplicated reactor trip as a result of operating closer to reactor trip setpoints. An uncomplicated reactor trip, however, is not an accident described in the FSAR. 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 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 FSAR? ☐ Yes ☒ No

BASIS:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR. The following is the basis for concluding that the proposed change will not affect the FSAR-described design function of ICI assemblies, CETs, HJTCs, COLSS, and CECOR.

A modification of the ICI Quickloc Assemblies (ER-W3-2003-0135-00) will be implemented to accommodate possible excessive ICI thimble growth which can cause the bottom of the thimbles to rest on the bottom of the fuel guide tube. The modification will consist of modifying the Quickloc Seal Carrier Assembly by installing a spacer in the seal carrier to elevate the ICI Stalk Assembly Plug to provide more bottom clearance to accommodate ICI thimble growth.

The purpose of the spacer is to maintain the original design function of the seal carrier assembly while accommodating ICI thimble growth. The instrument assemblies perform a passive safety related function by providing an RCS pressure boundary. This modified configuration will still meet the original design and ASME Section III Class 1 requirements. The design function of the seal carrier assembly will not be changed. Installation of the spacer will ensure that the original design function of the assembly will be met in the event of ICI thimble growth. Therefore, this change will have no impact on the design function of the assembly.

The potential increase in the vertical height of ICIs does not affect the other design function as described in the FSAR or other License Basis Documents (LBDs). The function of CETs is not affected by this change. According to Westinghouse document DAR-ME-03-24, in the worst case of 1.5 inch upward movement of the incore instrument, the CETs will remain between flow slots within the ICI thimble, ensuring that the sensed temperature at the CETs will not change. Since the CETs are used by the operators to monitor core conditions during an accident and are only measuring bulk core exit temperatures, a change in vertical height of up to 1.5 inches is not significant and does not affect the function of the CETs to allow operators to monitor core exit temperatures. HJTCs are used by the operators during an accident to determine level in the upper plenum and head. Exact level indication is not needed (or possible, given the crude nature of HJTC function). A possible 1.5 inch increase in the HJTC elevation is conservative in that the EOP actions initiated as a result of HJTC indications would occur slightly earlier. Therefore, the HJTCs will continue to provide their design function of guiding the operators to take appropriate action during an accident. The CETs and HJTCs also provide input to the saturation monitor, but a 1.5 inch change in the elevation of these instruments would have a negligible effect on the determination of subcooling, since the upper plenum temperature (that is input to the saturation calculation) is a bulk temperature not affected by a small change in elevation within the ICI shroud.

(continued)

BASIS (Continued):

Both COLSS and CECOR will be updated to reflect the changed geometry of the FIDs (incore detectors). This will ensure that these programs continue to accurately represent the core and perform their intended function. In the interim before the COLSS constants are updated, conservative penalty factors will be added to COLSS to ensure that it will continue to perform its design function, described in the FSAR, of continuously monitoring limiting conditions for operation (LCOs) for PLHGR, DNBR margin, and core power (FSAR 7.7.1.5.1), and alarm if necessary to alert the operators if the LCOs are not being met. Since CECOR will be updated to model the actual ICI positions following addition of spacers before its use in startup testing or core monitoring, it will be able to accurately calculate power distribution even in the interim before the COLSS setpoint analysis is completed.

An axial shift in ICI positions could affect the following COLSS algorithms: DNB power operating limit (POL), LPD POL, ASI monitoring, and azimuthal tilt monitoring. The effect of ICI axial shift on the POLs and ASI in COLSS will be accomplished via addition of an adjustment to the COLSS EPOL terms. (In the long term, the ASI impact may be accommodated via either a revised ASI range in the COLR or adjustments to the COLSS ASI-related constants; in either case, COLSS will continue to be able to perform its design function.) Westinghouse estimates that in the interim before the COLSS setpoint analysis is redone, a 4-5% penalty on thermal margin will result from the addition of a conservative EPOL adjustment. This adjustment will cover ICI shift-induced uncertainty in the POL and ASI calculations in COLSS. After the setpoint analysis is completed, the long-term adjustment on EPOL is expected by Westinghouse to be about a 1-2% impact on thermal margin. According to Westinghouse and Waterford Reactor Engineering, the expected thermal margin without the ICI spacer mod is in the range of 10-11%. Thus, even the larger interim EPOL penalty will preserve sufficient margin to safely operate the reactor at full power. Since COLSS only provides assistance to the operators in maintaining the thermal margin-related tech spec LCOs, a reduction in margin is not a malfunction of COLSS (it still conservatively performs its design function).

An axial shift in ICI positions could also potentially affect azimuthal tilt monitoring in COLSS and the tilt allowance input into CPC. Since CECOR is used to maintain the COLSS tilt calculation conservative, and CECOR will accurately calculate tilt, COLSS tilt will be maintained conservative. CPC tilt is made conservative with respect to COLSS, so since COLSS tilt will be maintained conservative by the use of CECOR, CPC tilt will also be maintained conservative.

An additional potential impact of an ICI axial shift is on radial peaking factor (RPF) surveillance, CISAM (excore detector shape annealing matrix for input to CPCs, determined during startup testing), and RCS flow rate surveillance. Since these measurements use CECOR to determine core power distribution, and CECOR will be changed to reflect the new ICI geometry before use, an ICI axial shift will not affect these measurements.

Finally, there is no impact on fuel misloading detection from a shift in the axial location of the ICIs, because for fuel misloading we must be able to detect *radial* asymmetry, which is not affected by the axial shift.

In conclusion, an axial shift in ICI positions will not cause more than a minimal increase in the likelihood of malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

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

BASIS:

An impact on the consequences (radiological dose) of an accident would require a change in the way the plant responded to an accident initiator. Most of the impact of the proposed addition of ICI spacers is related to COLSS thermal margin, as evaluated for Question 2. Since thermal margin is only related to maintenance of certain tech spec LCO conditions, pre-accident, changes in thermal margin could only affect accident consequences if the reactor protection system were unable to perform its design function in response to an accident initiator. But COLSS will be adjusted via EPOL penalty factor to ensure that the tech spec LCO is maintained and, thus, by definition, the reactor protection system will be able to provide its design function and there will be no impact on accident consequences.
In addition, the potential increase in the vertical height of ICIs does not significantly affect the design function of the inadequate core cooling instruments (ICCI), i.e., CETs, HJTCs, and saturation margin. As described for Question 2, these instruments will still perform their design function of providing information to the operators to guide their response to an accident within the context of the emergency operating procedures. Therefore, the change in the axial location of the ICCI components will not affect the consequences of an accident.

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 FSAR? ☐ Yes ☒ No

BASIS:

The proposed changes do not affect any structures, systems, or components the failure of which would affect the consequences of an accident. The main effect of the proposed change is to conservatively reduce the thermal margin during power operation. As discussed for Question 2, the design function of systems affected by the change is maintained, so there is no impact of the change on the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

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

BASIS:

The change to the ICI assemblies of adding a spacer can not create the possibility for an accident of a different type because failure of an instrument penetration is already evaluated in the FSAR. The modified ICI configuration will still meet the original design and ASME Section III Class 1 requirements. The design function of the seal carrier assembly will not be changed. Installation of the spacer will ensure that the original design function of the assembly will be met in the event of ICI thimble growth. In addition, use of a penalty factor in COLSS can not create the possibility of a different accident since the purpose of the additional penalty is to maintain the design function of COLSS to assist the operators in monitoring tech spec thermal margin LCOs.

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 FSAR? ☐ Yes ☒ No

BASIS

The proposed change does not create the possibility of a malfunction with a different result because the change is to the ICI seal assembly, which is already evaluated in the FSAR. No new interactions with other structures, systems, or components are created, nor is the potential for a different failure mode for the ICI seal assembly created, since the addition of spacers only moves the ICI assembly a small amount within the instrument nozzle. The addition of a COLSS penalty factor does not create the possibility of a malfunction with a different result since the purpose of the additional penalty is only to maintain the design function of COLSS to assist the operators in monitoring tech spec thermal margin LCOs.

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

BASIS:

The proposed change is to a part of the reactor coolant system boundary, which is a fission product barrier. The modified ICI configuration will still meet the original design and ASME Section III Class 1 requirements. The design function of the seal carrier assembly will not be changed. Installation of the spacer will ensure that the original design function of the assembly will be met in the event of ICI thimble growth. Therefore, there is no effect on the integrity of this fission product barrier. Another fission product barrier is the fuel cladding. COLSS does not directly affect the response of the NSSS to an accident that could affect fuel cladding. COLSS is used to ensure that the core is operated with sufficient thermal margin that fuel cladding damage will not occur for anticipated operational occurrences, and that the degree of cladding damage in more severe events remains bounded by the safety analysis described in the FSAR. The addition of the COLSS penalty will ensure that the required thermal margin is maintained during operation to protect the fuel cladding as evaluated in the FSAR. Therefore, the addition of the COLSS penalty will not result in the cladding fission product barrier being affected.

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

BASIS:

None of the COLSS or CPC tilt adjustments represent a change to a methodology described in the FSAR. These adjustments are at a level of detail that is below what is described in the FSAR and are part of the normal process of incorporating instrument uncertainties into COLSS and CPC. The use of EPOL penalties is part of the current COLSS method of accounting for instrument uncertainty; the increased penalty is simply an application of this existing method to a larger instrument uncertainty resulting from the ICI axial position change. In the case of ASI, adjustment of the COLR ASI range or the COLSS ASI-related constants follows the existing method of evaluating the impact of core changes on ASI monitoring.

CECOR, which provides key core monitoring information for COLSS and CPC adjustment (e.g., with respect to azimuthal tilt) will have its database constants adjusted to reflect the changed ICI geometry. This does not affect CECOR as described in the FSAR (the description in the FSAR is not to the level of detail of ICI geometry); the geometry change, moreover, is not a methodology-related change, but only a change in physical geometry input. (Although code input changes can be methodology changes under 50.59, input changes that represent changes to methodology are changes to such key inputs as heat transfer coefficients or DNB correlation constants, and must be described in the FSAR.) Therefore, the proposed changes to COLSS or CECOR do not represent changes to a method of evaluation described in the FSAR.

I. OVERVIEW / SIGNATURES

Facility: Waterford-3
Document Reviewed: ER-W3-2000-0106-002 **Change/Rev.** 0

System Designator(s)/Description: RC-Reactor Coolant, CVC-Chemical and Volume Control System

Description of Proposed Change

The Design Conditions for the piping directly downstream of valve CVC-403 up to and including throttle valve CVC-4061 and bypass valve CVC-4064 in the Controlled Bleedoff System, needs to reflect the maximum operating pressure of 120 psig and design pressure of 210 psig for this identified portion of the system. The existing documented operating and design pressures are 50 psig and 75 psig respectively.

The relief valve as presently installed serves no over pressurization function as the design function is to allow a flow path for controlled bleedoff in the event a downstream valve is positioned closed. Over-pressurization protection will be credited as a result of the changes proposed whereas the subject piping will be protected from pressures in excess of the calculated 204.3 psig pressure resulting from the 150 psig relief valve setpoint. The present administrative open control of block valve RC-602, will also be credited in the provision for over pressure protection with relief valve RC-603 therefore ensuring the proposed 210 psig system design pressure will not be exceeded. Since Waterford 3 is committed to design and construct systems in accordance with ASME Section III requirements, this crediting of the existing administrative control to ensure valve RC-602 remains open is considered a deviation from the original plant licensing commitments. Due to the Waterford-3 construction permit issuance prior to 1984, this code exception may be evaluated under the provisions of 10CFR50.59 (Reference letter dated 9-17-97 from NRC to Entergy in Response to Request for Relief from 1971 ASME Boiler & Pressure Vessel Code Section III, Paragraph NC-7153). No additional operational controls or testing is required for valves RC-602 and RC-603 to provide the proposed system over pressure protection.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: -04-001 -04-001	Sections I, II, III, IV, and VI required

Preparer: Thomas R. Hempel / Thomas R. Hempel / EOI / Design Engineering Mechanical / 1-21-04
Name (print) / Signature / Company / Department / Date

Reviewer: John Russo / John Russo / EOI / Design Engineering Mechanical / 1-21-04
Name (print) / Signature / Company / Department / Date

OSRC: K. Peters / 1/29/04
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name: **Scope of Assistance:**

None

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	FSAR Table 3.2-1, (DRN 03-759); FSAR Chapter 9 (DRN 03-760)
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. **Basis**

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The Licensing Research System (LRS) Entergy-Fulfind electronic search was utilized for LBD applicability determination as described in the Reference Section C below. The following FSAR sections were considered for applicability to the changes proposed. FSAR Table 3.2-1 Classification of Structures, Systems and Components, lists the critical characteristics of the Chemical and Volume Control System with clarifying notes. DRN 03-759 revises this table to include existing Note 28 which refers to administratively controlled block valves in the piping for pressure relieving devices as part of the RCPB (reactor coolant pressure boundary). FSAR Section 5.4 Component and Subsystem Design, and Section 9.3.4.1.1 describe the Reactor Coolant Pumps Controlled Bleedoff (CBO) System, pump seal assemblies design basis and functional requirements. DRN 03-760 revises FSAR Section 9.3.4.3.2 Overpressure Protection, to revise the statement for the Reactor Coolant Pump Controlled Bleedoff Header Relief Valve to delete "it does not serve an overpressure protection function". Other License Basis Documents identified by the electronic search such as the Safety Evaluation Report (SER), and Technical Requirements Manual (TRM) were reviewed but found unaffected by the changes proposed involving revision to design and operating pressures, in addition to crediting existing relief valve RC-602 for overpressure protection for the identified portion of the CBO system. The proposed changes will not require testing because the pressure revision is acceptable and in compliance with all applicable ASME and ANSI codes for all of the materials used in the subject valves and piping application while the relief valve RC-603 and block valve RC-602 are currently tested under the Waterford 3 Inservice testing (IST) program which provides assurance of proper valve operation.

C. **References**

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
Using WF3 Fullfind database search of the LBDS_50_59	The keywords "controlled" and "bleedoff" were utilized producing 10 instances requiring investigation to determine potential impact.

LBDs/Documents reviewed manually:

FSAR Sections: 1.9, 3.9, 5.4, 7.5 and 9.3 FSAR Tables: 3.2-1, 6.2-1, and 9.3-15 Technical Requirements Manual: 3/4.6 Technical Specification: 3/4.4 NUREG 0787 Safety Evaluation Report: 5.2, 9.3	ER-W3-2000-0106-000 CBO Backpressue Control – 50.59 Evaluation # 01-025

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
☒ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) _____ ☐ No

_____

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

The design and operating pressure for the portion of Controlled Bleedoff System piping between valves CVC-403, CVC-4061 and CVC-4064 requires revision to reflect the approved maximum system operating pressure. This identified section of pipe may be subject to pressure up to the operating limit of 120 psig due to the automatic or manual throttling capability of the system provided by ER-W3-2000-0106-000. Over-pressure protection of this identified piping section can be credited by existing relief valve RC-603 (150 psig setpoint), with the administratively controlled normally locked open in-line block valve RC-602. Relief Valve RC-603 is currently designed to provide an alternate flowpath for Reactor Coolant Pump Controlled Bleedoff in the event of a downstream valve closure. The design criteria in the FSAR states this relief valve does not serve an overpressure protection function which will require revision by the change proposed.

Reason for proposed Change:

Condition Report No. WF3-2002-0845 identified that plant modification ER-W3-2000-0106-000 which added an automatic pressure regulating valve and manual throttling bypass valve, failed to consider the most limiting allowable Controlled Bleedoff system operating pressure during plant startup or shutdown of 120 psig for the piping directly upstream of these new valves.

50.59 Evaluation summary and conclusions

This evaluation included a review of the Licensing Basis Documents identified in the Section II-A Screening, and an Entergy-Fulfind electronic search using the keywords identified in Section II-C. The evaluation determined the frequency of an accident or Structure, System or Component malfunction previously evaluated in the Safety Analysis Report (SAR) will not be increased. The affected portions of the Reactor Coolant Pumps Controlled Bleedoff System are not considered an accident initiator. The evaluated subject change to revise the design and operating pressure for the portion of the Controlled Bleedoff system directly upstream of the pressure control and throttle valves, from 75 / 50 psig to 210 / 120 psig respectively is within acceptable code (ASME / ANSI) pressure limitations for the materials installed. These design and operating pressure values are not depicted in any of the license basis documents therefore no revisions of these documents are necessary to support this facet of the proposed change.

The proposed change of utilizing existing relief valve (RC-603) for overpressurization prevention along with crediting the current administrative control which ensures in-line block valve RC-602 remains open is considered an alternative to the ASME Section III Code requirements for overpressure protection. The regulations in 10CFR50.55a(d) concerning meeting the ASME Code requirements do not apply since Waterford 3's construction permit was docketed prior to May 14, 1984. This administrative control code exception has therefore been evaluated as a license basis change allowed under 10CFR50.59. The changes proposed will not increase the consequences of an accident or SSC malfunction, and the change does not create the possibility of an accident or SSC malfunction of a different type than previously evaluated in the SAR. No new system interactions, design basis limit for a fission product barrier or new design basis analysis is required or created as a result of the changes evaluated. There is no change to any Technical Specification, and prior approval by the Nuclear Regulatory Commission is not required. Several Final Safety Analysis Report (FSAR) changes are necessary to describe the new over pressure protection feature of valve RC-603, and the administrative control of valve RC-602, as shown on the DRN's listed in Section II-A.

B. License Amendment Determination

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 FSAR? ☐ Yes ☒ No

BASIS:

FSAR Section 15 "Accident Analysis" identifies accidents previously evaluated in the FSAR. This chapter was reviewed and determined that the frequency of occurrence of a "Decrease in Reactor Coolant Accident (LOCA) is the only accident which could potentially be impacted by the proposed changes. The LOCA potential was reviewed because relief valve RC-603 and associated block valve RC-602 are located on portions of the Controlled Bleedoff System classified as Reactor Coolant Pressure Boundary which in accordance with ANSI N18.2a-1975 indicates the RCS extends to the outermost containment isolation valve.

This evaluation has determined that the frequency of a LOCA is not increased. The existing valves functions will be credited with providing overpressure protection to the identified portion of the Controlled Bleedoff System. The changes proposed will not require revising the normal position of any of the subject valves. These existing valves are already encompassed by the present LOCA analysis which remains unaffected by the changes proposed under this ER and therefore does not increase the frequency of an accident described in the SAR.

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 FSAR? ☐ Yes ☒ No

BASIS:

The proposed changes will not increase the likelihood of occurrence of a malfunction of a SSC important to safety. The changes are inclusive to the Chemical and Volume Control Reactor Coolant Pump Controlled Bleedoff System which is designed to the requirements of ASME Section III, Class 2. The subject changes proposed do not require any physical changes to the system therefore all of the system characteristics and design functions will remain unchanged other than crediting valve RC-603 with providing over pressure protection. Revising the design and operating pressure for the identified portion of the system directly upstream of the throttle and pressure regulating valves conforms to the existing system operating parameters. The materials and components installed in the identified piping section have been found in accordance with the applicable ASME / ANSI codes for the revised 210 psig design pressure at the system design temperature of 250 degrees F.

The crediting of existing relief valve RC-603 along with the administrative controls to ensure in-line block valve RC-602 remains open, will provide over pressure protection for the identified piping section ensuring the subject revised design pressure is not exceeded. Existing relief valve RC-603 and block valve RC-602 will remain part of the Inservice Test Plan (IST) and no increase in the likelihood of occurrence of a malfunction can be attributed to crediting these valves with over pressure protection of the piping section described. Crediting over pressure protection by the use of existing valves does not require a piping system configuration change therefore no impact of system redundancy or independence is created.

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

BASIS:

The FSAR Chapter 15, Accident Analysis was reviewed with the only accident consequences (radiation dose) that the proposed change could possibly impact by association with the Reactor Coolant System is a Loss of Coolant Accident (LOCA) as described in Subsection 15.6.3.3. Since the proposed change to the Reactor Coolant Pump Controlled Bleedoff System will credit an existing valve RC-603 with overpressurization protection of the identified piping section, the system flow path remains unchanged without creation of any new release path therefore the previous LOCA analysis will be unaffected. The effect of Station Blackout (SBO) has also been evaluated in regard to the identified piping section between valves CVC 403, CVC 4061 and CVC 4064, without impact. Since the Containment Isolation valves RC 606 and CVC 401 are fail closed valves, the identified section of piping located outside of the containment will not see the increase in pressure which results from the postulated seal failure of all the Reactor Coolant Pumps during the Station Blackout scenario.

An Interfacing System LOCA was also considered without impact by the subject changes proposed. The setpoint (75 psig), and capacity (250 gpm), of the Volume Control Tank relief valve CVC 182 will prohibit pressure for the identified section of the Control Bleedoff system in excess of design pressure resulting from a postulated instantaneous Reactor Coolant Pump seal failure.

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 FSAR? ☐ Yes ☒ No

BASIS:

The proposed changes associated with revising the design and operating pressure for an identified portion of the Controlled Bleedoff system to conform to the previously approved existing system operating pressure, does not have malfunction consequences association because the *materials in the application are suitable for the revised design and operating pressures.*

The consequences of a malfunction associated with the failure of subject relief valve RC-603 to open has been previously considered in FSAR Table 9.3-15 Failure Modes and Effects Analysis. The effects from a loss of bleed-off header over pressure protection which are denoted as no impact on the normal system operation, remain unchanged as a result of the proposed changes.

Failure of pressure regulator valve CVC-4063 to open could also result in pressurization of the subject Controlled Bleedoff header to the setpoint of relief valve RC-603. This malfunction would be identical to the fail close characteristic of existing and previously evaluated Containment Isolation valves CVC-401 and RC-606. Pressurizing the header to the setpoint of the relief valve is a previously evaluated system design feature that does not increase the consequences of the Controlled Bleedoff system or component malfunction by the changes proposed.

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

BASIS:

The changes proposed under ER-W3-2000-0106-002, do not require any physical plant or system changes, or require revision to the subject Controlled Bleedoff System present operation. Crediting the existing plant configuration without any new system interface will therefore not create an accident of a different type or new failure modes other than previously evaluated in the FSAR.

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

BASIS

The changes proposed cannot create the possibility of a malfunction of a SSC important to safety with a different result than previously evaluated in the FSAR because no new component will be added, nor will the existing components operate in a manner different than previously approved in the Controlled Bleedoff System. Using existing relief valve RC-603 with its existing setpoint enables the system to operate as previously designed without impact from the proposed change which credits this valve in protecting the identified portion of the system upstream of the pressure regulator valve and throttle valve.

Failure of backpressure control valve CVC-4063 to open with the bypass valve CVC-4061 closed could cause the controlled bleedoff header pressure to increase to the 150 psig set point of relief valve RC-603. Flow would then be directed through the relief valve to the Quench Tank per the existing previously approved and evaluated system design. The failure of CVC-4063 to open would be identical to the fail close characteristic of containment isolation valves CVC-401 and RC-606 also located in the subject Controlled Bleedoff header. Therefore no new component malfunction with a different result than previously evaluated in the FSAR can be realized.

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

BASIS:

The proposed change associated with revising the design and operating pressure for a portion of the Controlled Bleedoff System between valves CVC-403, CVC-4064 and CVC-4061 to meet the previously approved maximum system operating pressure (OP-002-005), as well as the crediting the over-pressure protection of existing relief valve RC-603 with the administrative control of block valve RC-602, are changes that do not impact any fission product barriers such as fuel cladding, reactor coolant system boundary, or containment. The proposed change does not affect any numerical analytical limits used in the design process.

The function and abilities of the subject Controlled Bleeedoff header containment isolation valves CVC-401 and RC-606 to provide isolation is not impacted by the changes proposed.

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

BASIS:

The subject proposed changes to reflect the approved Controlled Bleedoff system maximum operating pressure in the design conditions for the piping between valves CVC-403, CVC-4061 and CVC-4064, in addition to crediting existing relief valve RC-603 along with the administrative control to ensure in-line valve RC-602 remains open, will not require re-analysis of any existing design basis or safety analyses. The aforementioned changes proposed result in the Chemical and Volume Control System Reactor Coolant Pump Controlled Bleedoff to function identically as previously approved. All intended system design functions, design basis limits of fission product barriers and accident consequences are unaffected by the changes proposed.

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 Steam Electric Station

Document Reviewed: ER-W3-2003-0755-000 **Change/Rev.:** 0

System Designator(s)/Description: Steam Bypass Control System (SBCS)

Description of Proposed Change

The proposed change, implemented per Engineering Request ER-W3-2003-0755-000, revises the Waterford 3 licensing and design bases to indicate that the reactor may trip on a turbine trip between 50% and 70% power at certain times in core life. The current licensing and design bases state that the Steam Bypass Control System (SBCS) is designed to prevent a reactor trip on a turbine trip. The proposed change involves changes to Waterford 3 Final Safety Analysis Report Chapters 7 and 10 and Waterford 3 Main Steam Design Basis Document (W3-DBD-006). The issue was first identified by Combustion Engineering (now Westinghouse) in 1993 and then again in 2003 during work performed for the Extended Power Uprate Project.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: ____) 04-002	Sections I, II, III, IV, and VI required

Preparer: R. T. Finch / R. T. Finch EOI / Design Engr. / 2/26/04
Name (print) / Signature / Company / Department / Date

Reviewer: D. J. Rohli / D. J. Rohli EOI / Design Engr. / 2/26/04
Name (print) / Signature / Company / Department / Date

OSRC 3-3-04
Chairman's Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.9) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
NA	NA

II. SCREENING

A. Licensing Basis Document Review

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☒	☐	Sections 7.4.2.5, 7.7.1 (DRN-03-2155) and Sections 10.3.1, 10.4.4 (DRN-03-2157)
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☒	☐	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculation Manual[3]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ Yes ☒ No
If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation? ☐ Yes ☐ No ☒ N/A
(Check "N/A" if dry fuel storage is not applicable to the facility.)
If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
(See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

[1] If "YES," see Section 5.1.4. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. **Basis**

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The Waterford 3 Final Safety Analysis Report (FSAR) contains the statement that the SBCS design "Permits the accommodation of load rejections of any magnitude without opening either the pressurizer or the steam generator safety valves or tripping the reactor" (reference FSAR section 7.7.1.4.1.a). It also contains the statements "If a large rapid reduction in power demand occurs, the SBS bypasses steam to the condenser to prevent tripping of the reactor (reference FSAR section 10.3.1) and "The Steam Bypass System is designed to accomplish the following functions: a) to permit turbine load rejections to the condenser up to 65 percent of the rated main steam flow without opening the main steam safety valves or tripping the reactor" (reference FSAR section 10.4.4.1). The Waterford 3 Safety Evaluation Report (SER) contains these same types of statements in Sections 10.1 and 10.4.4. ABB Combustion Engineering Nuclear Power report titled, "Final NSSS Control System Setpoint Study for Operation at Reduced RCS Temperatures", (ABB-CE Doc. No. C-MECH-93-025) states that the current setpoints for the Steam Bypass Control System could result in a Reactor Trip if the Turbine were to trip between 50% and 70% power at certain times in core life. This is in conflict with the FSAR / SER statements described above. The Engineering Request Review Group (ERRG) decided to not pursue a plant modification to correct the issue. The FSAR thus must be changed to correctly reflect plant operation. The Operating License / Technical Specifications are not impacted by the proposed change because these documents do not contain a discussion of the Steam Bypass Control System.

The proposed change is strictly a paper change to the design (DBD-006) and licensing (FSAR) bases. No physical changes are being made to the plant and therefore no testing or experimentation is required.

Waterford 3 does not have an Independent Fuel Storage Installation.

C. **References**

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
LBDS_50_59	"steam bypass", "SBS", "SBCS", "reactor trip", "anticipated operational occurrence"

LBDs/Documents reviewed manually:

FSAR Sections 3.1.6, 3.1.11, 3.1.16, 3.1.24, 3.1.25, 4.2.1, 7.2, 7.4.2.5, 7.7.1.4, 10.3.1, 10.4.4, 15.0, 15.1.1.3, 15.2.1.1, 15.2.1.2 and 15.2.3.2. ABB Combustion Engineering Nuclear Power Report, "Final NSSS Control System Setpoint Study for Operation at Reduced RCS Temperatures", April 23, 1993, ABB-CE Doc. No. C-MECH-93-025. CR-WF3-2003-1363 and CR-WF3-2003-2316. ER-W3-2003-0352-000. Procedure Nos. OP-902-002, "Loss of Coolant Accident Recovery" and OP-902-003, Loss of Offsite Power / Loss of Forced Circulation Recovery". SER Sections 10.1 and 10.4.4. Calculation No. EC-S93-008 Rev. 2, "Study Calc. – Waterford 3 Level 1 Internal Events PSA Model".

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☐ Yes
☒ No

If "Yes," list the required changes.

NA.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ☐ No

Name of Security Plan reviewer (print) / Signature / Date
NA

VI. 50.59 EVALUATION

A. <u>Executive Summary</u> (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

The proposed change is to revise the Waterford 3 Design and Licensing Basis to identify that the reactor may trip following a turbine trip at power levels between 50% and 70%.

Reason for proposed Change:

ABB Combustion Engineering Nuclear Power report, "Final NSSS Control System Setpoint Study for Operation at Reduced RCS Temperatures", April 23, 1993, (ABB-CE Doc. No. C-MECH-93-025) concludes that the current setpoints for the Steam Bypass Control System could result in a Reactor Trip if the Turbine were to trip between 50% and 70% power at certain times in core life.

50.59 Evaluation summary and conclusions

The proposed change is to the design and licensing bases of the non-safety Steam Bypass Control System. There are no physical changes to the plant. The change has no impact on the frequency of occurrence of an accident or of the consequences of any accident analyzed in the FSAR. The proposed change does not create the possibility of a new type of accident, does not impact a fission product barrier and does not involve a new method of evaluation. The change does minimally increase the probability of a reactor trip which is categorized as an anticipated operational occurrence and not an accident. The proposed change therefore does not require prior Nuclear Regulatory Commission approval.

B. License Amendment Determination

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 FSAR?

☐ Yes
☒ No

BASIS:

The potential for the reactor to trip following a turbine trip at power levels between 50% and 70% at certain times in core life is not addressed in the FSAR because when the FSAR was written it was thought that the design of the steam bypass control system would preclude this type of reactor trip. The proposed change is to revise the design and licensing basis to describe the current operation of the SBCS.

A turbine trip is a FSAR anticipated operational occurrence (AOO - Reference FSAR Section 7.2.2.1.1). FSAR Section 15.2.1.1 "Loss of External Load" evaluates the event of a turbine trip with the Steam Bypass System in the manual mode. The proposed change could result in an event that is similar to this FSAR event in that if the SBCS failed to perform its function at power levels between 50% and 70%, at certain times in core life, the result would be the same as if the SBCS was unavailable to perform its design function, i.e. was in the manual mode and immediate operator action could not be credited. In the FSAR evaluated event the transient is terminated by a reactor trip on high pressurizer pressure. If the SBCS failed to perform it design function between 50% and 70% power the Reactor Protective System would in this case also trip the reactor on high pressurizer pressure. The proposed change does not change the probability of the FSAR accident initiator (the turbine trip) but it does increase the probability of a reactor trip because if the SBCS could not perform its design function a reactor trip is required to terminate the transient. FSAR Section 4.2.1.1 defines a Reactor Trip as an anticipated operational occurrence and not an "Upset" or "Accident" event. The plant has been designed so that no adverse consequences result from a reactor trip. It is estimated that the increased likelihood of a reactor trip due to the proposed change is $5.0 \times (10)^{-4}$ / year (Reference CR-WF3-2003-1363 Corrective Action No. 1, Attachment No. 1). This is a minimal increase in the probability of a reactor trip (the base probability of a reactor trip is $6.0 \times (10)^{-1}$ / year (Reference Calculation No. EC-S93-008 Rev. 2)).

FSAR Section 15.2.3.2 addresses a loss of normal feedwater flow with an active failure in the steam bypass system (stuck open bypass valve) but the probability of a bypass valve failing to close is not affected by the proposed change. The proposed change has no impact on the reliability of the bypass valves. The proposed change therefore has no impact on the FSAR section 15.2.3.2 event.

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 FSAR?

☐ Yes
☒ No

BASIS:

The Steam Bypass Control System is classified as non-safety. The proposed change is to the plant design and licensing basis only. There are no physical changes to the plant. As discussed in the answer to Question No. 1 above the reliability of the steam bypass valves is not impacted by the proposed change.

Although the SBCS is classified as non-safety it is credited, in procedures such as OP-902-002, "Loss of Coolant Accident Recovery" and OP-902-003, "Loss of Offsite Power / Loss of Forced Circulation Recovery", as a means to perform controlled cooldown (Reference Instructions No. 19 and No. 28 respectively in the two procedures) of the Reactor Coolant System during accident recovery. The proposed change does not impact the ability to use the SBCS for such actions because these actions are performed after the reactor has tripped.

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

BASIS:

As discussed above the SBCS is classified non-safety and as such is not credited with accident mitigation. In the event described in Westinghouse report No. C-MECH-93-025, where the SBCS fails to prevent a reactor trip, the reactor protective system trips the reactor on high pressurizer pressure. The proposed change does result in a challenge to the safety related reactor protective system that would not occur had the SBCS functioned as originally designed but this does not increase the consequences of any accident evaluated in the FSAR. As discussed in the answer to Question No. 1 above the probability of a reactor trip is increased minimally. As discussed above FSAR section 15.2.3.2 assumes a stuck open bypass valve and this is not impacted by the proposed change.

Also as discussed above the proposed change has no impact on the ability to use the SBCS to cool down the Reactor Coolant System per various Emergency Operating Procedures such as OP-902-002, "Loss of Coolant Accident Recovery" and OP-902-003, "Loss of Offsite Power / Loss of Forced Circulation Recovery".

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

BASIS:

As discussed above the Steam Bypass Control System is non-safety, the proposed change does not involve a physical change to any plant system, structure or component and the proposed change does not impact the ability of the SBCS to be used to cool down the Reactor Coolant System post accident. Also as discussed above the proposed change does result in a minimal increase in the probability that the Reactor Protective System will be required to trip the reactor but this event is in the design basis of the Reactor Protective System.

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

BASIS:

As described above, for the event described in Westinghouse report No. C-MECH-93-025, where the SBCS fails to prevent a reactor trip, the reactor protective system trips the reactor on high pressurizer pressure. This is not a new type of event. As discussed in the answer to Question No. 1 above the reactor protective system is specifically designed for this type of event.

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 FSAR? ☐ Yes ☒ No

BASIS

As described above the Steam Bypass Control System is non-safety and no physical changes are being made to any plant system, structure or component. Also, as discussed in the answer to Question No. 1 above, a reactor trip resulting from a turbine trip is the same event as discussed in FSAR Section 15.2.1.1.

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

BASIS:

As described above the proposed change to the non-safety steam bypass control system has no impact on the operation of the safety related reactor protective system. For the event described in ABB Combustion Engineering Nuclear Power report, "Final NSSS Control System Setpoint Study for Operation at Reduced RCS Temperatures", (ABB-CE Doc. No. C-MECH-93-025) the transient initiated by the turbine trip is terminated by a reactor protective system reactor trip generated on high pressurizer pressure. The Waterford 3 reactor protective system has been specifically designed to terminate this type of transient with no resultant degradation to any of the three fission product barriers, i.e. fuel clad, reactor coolant system pressure boundary and containment building.

American National Standard N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" defines a reactor-turbine load mismatch, including loss of load and turbine trip as a Condition II event (Incidents of Moderate Frequency). The design requirements for such events are "a single Condition II incident shall not cause consequential loss of function of any barrier to the escape of radioactive products". The FSAR states that this requirement is met by the plant protective system (Reference FSAR Section 3.1.6, Criterion 10 - Reactor Design). There is therefore no impact on a design basis limit for a fission product barrier.

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

BASIS:

The transient analysis computer code that was used for the analysis described in ABB Combustion Engineering Nuclear Power report, "Final NSSS Control System Setpoint Study for Operation at Reduced RCS Temperatures", (ABB-CE Doc. No. C-MECH-93-025) is the "LTC" computer code. This is the same computer code that is used for all of the Waterford 3 reload transient analyses. The proposed change therefore uses the same methods as used in existing analyses of record.

I. OVERVIEW / SIGNATURES

Facility: Waterford 3 SES

Document Reviewed: STI-04-WF3-0002-00 **Change/Rev.** 0

System Designator(s)/Description: Control Room Tracer Gas Test

Description of Proposed Change

This evaluation is on a Special Test Instruction (STI) for determining the control room unfiltered in-leakage to support the NRC's Generic Letter 2003-001. Specifically, the test will quantify unfiltered in-leakage in the Isolation, Recirculation and Pressurization emergency modes of operation. A small concentration of an inert tracer gas will be injected into the control room envelope and the unfiltered in-leakage will be determined based upon the rate of change in the tracer gas concentration.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2_____. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
■	50.59 EVALUATION (#: 04-003)	Sections I, II, III, IV, and VI required

Preparer: John E. Wilbur/ /Entergy-Waterford III/ Engineering/ 3-18-04
Name (print) / Signature / Company / Department / Date

Reviewer: John Russo/ John Russo /Entergy-Waterford III/ Engineering/ 3-18-04
Name (print) / Signature / Company / Department / Date

OSRC: Bradford L. Houston Bradford L. Houston 4/9/04
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
Peter Lagus	Tracer Gas testing procedures
Mark Louque	Environmental Screening (EV-117)

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	■	
TS	☐	■	
NRC Orders	☐	■	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☐	■	
TS Bases	☐	■	
Technical Requirements Manual	☐	■	
Core Operating Limits Report	☐	■	
NRC Safety Evaluation Reports[1]	☐	■	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	■	
Emergency Plan[2]	☐	■	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	■	
Offsite Dose Calculations Manual	☐	■	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ■ Yes ☐ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ■ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

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B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

II.A.1

This Special Test Instruction (STI) will align the main control room habitability systems in various modes of operation, as described in the FSAR, such that the tracer gas methodology can be utilized to determine the unfiltered in-leakage into the control room envelope. The various modes of operation and equipment line-ups for each of the test are described in the FSAR and do not operate the system outside the bounds described in the FSAR or Technical Specifications. This test does not impact the facility or a procedure as described in any of the Licensing Basis documents.

II.A.2

This STI will align the habitability systems in the various modes of operation, as described in the FSAR, such that the tracer gas methodology can be utilized to determine the unfiltered in-leakage into the control room envelope. The various modes of operation and equipment line-ups for each of the test are described in the FSAR. However, injecting small concentrations of a tracer gas (sulfur hexafluoride and nitrogen mixture) into the control envelope is not described in the FSAR; therefore this test does represent a test or experiment not described in the FSAR.

II.A.3

The Waterford 3 facility does not currently have dry fuel storage.

C. **References**

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:

TS 3/4.7.6.5 Control Room Isolation

TS 3/4.7.6.5 Bases

FSAR Table 3.2-1 Classifications of SSC's

FSAR 6.4 Habitability Systems

FSAR 6.5.1 ESF Filter systems

FSAR Table 7.3-5 Components Actuated on SIAS

FSAR 9.4.1 Control Room HVAC System

FSAR 9.4.3 RAB Ventilation System

FSAR 9.4.

FSAR 7.3 Engineered Safety Features

FSAR 12.3.2.1 Shielding, Design Objectives

FSAR associated tables and figures

TRM 3/4.7.13 Switchgear Ventilation

LBDs/Documents reviewed manually:

NONE

Keywords: Electronic search with LRS on "LBDS 5059" using:

Control Room Air Conditioning,

SIAS, Toxic Gas, Habitability, HVC, SVS, RAB Normal

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
■ No

If "Yes," list the required changes.

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III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	■	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	■	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	■	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	■	Increase the amount of thermal heat being discharged to the river or lake?
5.	■	☐	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	■	☐	Discharge any chemicals new or different from that previously discharged?
7.	☐	■	Change the design or operation of the intake or discharge structures?
8.	☐	■	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	■	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	■	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	■	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	■	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	■	Involve the installation or modification of a stationary or mobile tank?
14.	■	☐	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	■	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

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IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	■	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	■	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	■	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	■	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	■	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	■	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	■	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	■	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	■	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	■	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) _____ ☐ No

Name of Security Plan reviewer (print) / Signature / Date

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V. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A, above.

A. Check the applicable boxes below. If any of the boxes are checked, a 50.59 Evaluation is not required. If none of the boxes are checked, perform a 50.59 Evaluation in accordance with Section VI. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.6.1.1:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; AND

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; AND

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.6.1.2. Reference 50.59 Evaluation # _____ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.6.1.3. Reference: _____

☐ The proposed activity is controlled by another regulation per Section 5.6.1.4.

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10CFR50.59 Review Program Guidelines for guidance.

_____

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VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

This STI will align the habitability systems in the various modes of emergency operation, as described in the FSAR, such that the tracer gas methodology can be utilized to determine the unfiltered in-leakage into the control room envelope. A small concentration of an inert tracer gas (sulfur hexafluoride (SF6) and nitrogen) will be injected into the control room envelope and the unfiltered in-leakage will be determined based upon the rate of change in the tracer gas concentration. The tracer gas will be injected at the north emergency outside air intake by inserting a short length of tubing into the intake. The tube will prevent the proper operation of the fire damper, thus it will be rendered inoperable during the test. Downstream of the injection point, concentrations will be taken at existing pitot tube vent lock connections. Control room air samples will be taken at various locations within the actual control room envelope by test personnel.

Reason for proposed Change:

The NRC issued Generic Letter 2003-01 in June, 2003, requiring all nuclear power plants to submit information that demonstrates compliance with their current licensing basis, design bases, and applicable regulatory requirements; and that suitable design, maintenance and testing control measures are in place for maintaining this compliance. Furthermore, the NRC instructed licensees to utilize the ASTM E-741 testing methodology (tracer gas), or other acceptable surveillance methodology.

50.59 Evaluation summary and conclusions

The main control room habitability system is provided to assure that the operators can remain in the main control room and take effective actions to operate Waterford 3 safely under normal conditions and maintain a safe condition post accident, as required by General Design Criteria 19 of Appendix A to 10CFR50. This STI will align the habitability systems in the various modes of emergency operation, as described in the FSAR, such that the tracer gas methodology can be utilized to determine the unfiltered in-leakage into the control room envelope. The concentration of the tracer gas will be extremely low, such that there will be no impact on the operators. The test has no impact on fuel clad, reactor pressure boundary, or containment. No design basis limit for a fission product barrier as described in the FSAR will be exceeded or altered. The method of evaluation described in the FSAR (e.g. offsite dose calculations, control room habitability calculations, and toxic gas evaluations) is not altered. The STI may serve as input for future evaluations, but this STI collects data and does not change the method of evaluations.

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B. <u>License Amendment Determination</u>

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 FSAR?

☐ Yes
■ No

BASIS:

This STI gives guidance on performing a tracer gas test on the control room envelope to quantify unfiltered in-leakage. Specifically, a small concentration of tracer gas (sulfur hexafluoride and nitrogen mixture) will be injected into the control envelope and air samples will be taken to determine tracer gas concentrations at various times. The concentration of the tracer gas will be approximately 0.1 ppm, a factor of 10,000 times less than the OSHA threshold limit value of 1000 ppm, per the MSDS on the tracer gas. If the tracer gas cylinder malfunctioned and the entire contents emptied into the control room envelope, the envelope concentration would not exceed 3.5 ppm, a factor of 300 times less than the OSHA threshold limit value of 1000 ppm, per the MSDS on the tracer gas. Additionally, if the entire contents of the tracer gas cylinder emptied into the control room envelope, oxygen levels would remain well above the required levels for control rooms. As an additional precaution, oxygen levels will be monitored throughout the test. A review of UFSAR Chapter 15 reveals that the control room HVAC and habitability systems are not initiators for any described accidents or events. The main control room habitability system is provided to assure that the operators can remain in the main control room and take effective actions to operate Waterford 3 safely under normal conditions and maintain a safe condition post accident, as required by General Design Criteria 19 of Appendix A to 10CFR50. The functional capability of the main control room habitability system is maintained. Thus, performing the STI cannot result in any increase in 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 FSAR?

☐ Yes
■ No

BASIS:

This STI gives guidance on performing a tracer gas test, using sulfur hexafluoride, on the control room envelope to quantify unfiltered in-leakage. The inert tracer gas concentration is extremely small and will therefore have no effect on the control room equipment, habitability systems, or operators. This gas has commonly been used for tracer gas testing at many other sites and at Waterford 3 for condenser in-leakage with no detectable effects.

The equipment line-up for this test will replicate that which is described in the FSAR for the various modes of emergency operation (Isolation, Recirculation, and Pressurization mode). The equipment will not be operated outside the bounds of existing procedures or Technical Specifications. This line-up does not make the habitability equipment inoperable nor prevent the equipment from performing its safety function.

The tracer gas will be injected into the north emergency outside air intake, thus bypassing the Broad Range Gas monitors. Additionally, the Broad Range Gas monitors are not programmed to detect SF6, therefore the monitors will not be affected by this test. The habitability equipment provides cooling for important to safety equipment in the control room. The low concentrations of tracer gas will not alter the cooling capacity of the equipment nor will the charcoal efficiency in the emergency filtration units be affected; therefore, the functionality of the habitability equipment will be maintained. The fire damper at the north emergency outside air intake will be inoperable while the tracer gas is injected, but existing administrative controls will establish the appropriate compensatory measures during this portion of the test.

Thus, the STI will not result in any increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

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

BASIS:

This test will not require the plant to be operated outside the bounds of existing procedures or Technical Specifications. The test will not increase the dose to the control room operators or the public; therefore, this STI will not result in any increase in the consequences of an accident previously evaluated in the FSAR.

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

BASIS:

This test will not require the plant to be operated outside the bounds of existing procedures or Technical Specifications. The north outside air intake fire damper will be inoperable during tracer gas injection; existing administrative controls will establish the appropriate compensatory measures. All other systems will remain operable and capable of performing their safety function during this test. Should a malfunction of important to safety equipment occur during the test, credited redundant equipment will continue to be available. No credible failure scenario could result in increased dose consequences beyond that previously assumed, as it would be bound by single failure criteria. The test will not increase the dose to the control room operators or the public; therefore, this STI will not result in any increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

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

BASIS:

This test will not require the plant to be operated outside the bounds of existing procedures or Technical Specifications. The equipment line-up will replicate that which is described in the FSAR for the various modes of emergency operation (Isolation, Recirculation, and Pressurization mode). The north outside air intake fire damper will be inoperable during tracer gas injection; existing administrative controls will establish the appropriate compensatory measures. All other systems will remain operable and capable of performing their safety function during this test. No new system interactions or failure modes are created; thus, no possibility for an accident of a different type than any previously evaluated in the FSAR can 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 FSAR? ☐ Yes ■ No

BASIS

The equipment line-up for this test will replicate that which is described in the FSAR for the various modes of emergency operation (Isolation, Recirculation, and Pressurization mode). The north outside air intake fire damper will be inoperable during tracer gas injection; existing administrative controls will establish the appropriate compensatory measures. The low concentrations of tracer gas will not alter the cooling capacity of the equipment nor will the charcoal efficiency in the emergency filtration units be affected; therefore, the functionality of the habitability equipment will be maintained. All other systems will remain operable and capable of performing their safety function during this test. The tracer gas will be injected into the emergency outside air intake, thus bypassing the Broad Range Gas monitors. Additionally, the Broad Range Gas monitors are not programmed to detect SF6, thus the monitors will not be effected by this test. Thus, the STI will not result in any increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

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

BASIS:

The control room envelope and habitability systems has no impact on fuel clad, reactor pressure boundary, or containment other than providing a safe environment for the SSC's within the control room envelope. The low concentration of tracer gas inside the control room envelope will not have any effect on the operator's health or ability to perform their duties during normal or emergency operations. The control room habitability system will remain functional and will therefore maintain the required temperature for the equipment located within the envelope. Thus, the STI cannot result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

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

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BASIS:

This STI is being used to validate design information that provides basis for the control room habitability. Additionally, the tracer gas test has been reviewed and is required by the NRC's Generic Letter 2003-01. The method of evaluation described in the FSAR (e.g. offsite dose calculations, control room habitability calculations, and toxic gas evaluations) is not altered. The test may serve as input for future evaluations, but this STI collects data and does not change the method of evaluations; thus, does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

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