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Waterford 3

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

W3F1-2012-0034

April 30, 2012

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

Subject: Report of Facility Changes, Tests and Experiments and Commitment Changes for two year period ending April 25, 2012
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) and 10CFR72.48 (d)(2). This report covers the period from April 28, 2010 through April 25, 2012. This submittal also includes copies of the 10CFR50.59 Evaluations and 10CFR72.48 Evaluation from the summary report of the facility changes, tests and experiments for Waterford 3 and a summary report of Commitment Changes for the same time period in line with guidance in SECY-00-0045 and NEI 99-04.

If you have any questions regarding this report, please contact William J. Steelman at (504) 739-6685.

There are no new commitments contained in this submittal.

Sincerely,

WJS/MEM/

Attachment: Summary of Evaluations
Enclosure: Waterford 3 10CFR 50.59 and 10CFR72.48 Evaluations and Commitment Change Summary Report

IE47
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Attachment

W3F1-2012-0034

Summary of Evaluations

10CFR50.59 Evaluation Number	Initiating Document	Summary
10-01	EC-0000004019 R000	Zinc Injection Skid Addition
10-06	EC-0000014765 R000	SI-405A(B), Shutdown Cooling Suction Isolation Valves (Bypass Fill / Equalization Line Addition)
10-12	EC-0000018232 R001	Temporary Alteration to Maintain Fuel Pool Cooling Pumps Control Circuit Power Energized with SUPS AB De-energized
11-02	EC-0000025944 R000	Shutdown Cooling System Gas Accumulation Calculation Revision Reflecting Latest NRC/NEI Criteria for Management of System Gas Accumulation
11-03	EC-0000026496 R000	Safety Injection Sump Strainer Design Basis Revision Addressing the Single Failure of a Low Pressure Safety Injection Pump to Automatically Stop on a Recirculation Actuation Signal
11-04	OP-903-115 R019	Evaluation Crediting Manual Operator Action to Restore Reactor Coolant System Shutdown Cooling System During Integrated Emergency Diesel Generator/Engineering Safety Features Testing on Train A

10CFR72.48 Evaluation Number	Initiating Document	Summary
11-01	Calculation HI-2104606, Missile Calculation	Evaluation Reconciling Missile Calculation Analysis for Waterford HI-STORM CFSAR to Waterford 3 UFSAR for Intermediate and Small Missiles

Enclosure to W3F1-2012-0034

Waterford 3 10CFR50.59 and 10CFR72.48 Evaluations and Commitment Change Summary Report

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
2010-0003	P-23468	In response to Violation 96-07-01, for Control Room Envelope (CRE) inadequate post maintenance testing after airlock door seal replacement, Waterford 3 committed to performing Control Room Envelope (CRE) differential pressure testing demonstrating CRE integrity following airlock door maintenance. Waterford 3 added an option for post maintenance testing of performing alternate appropriate post maintenance testing demonstrating no adverse impact to CRE integrity and CRE airlock doors for certain types of airlock door maintenance (i.e. maintenance on Locksets, Mortise Cylinders/Cores and Trim Components).	Engineering has provided input that once certain types of door maintenance are completed (i.e. maintenance on Locksets, Mortise Cylinders/Cores and Trim Components) and components/ subcomponents are reinstalled, that an alternate industry based acceptable post maintenance smoke test of the areas worked is equally acceptable.
2010-0004	A-27220	Waterford 3 will submit a description of the MRP-227-a reactor vessel internals degradation management program, also referred to as an aging management program (amp), including the inspection plan, to the NRC for review and approval within 24 months after the final EPRI MRP recommendations are issued or within five years from the date of issuance of the uprated license, whichever comes first. Waterford 3 changed the commitment to make this submittal prior to December 31, 2011.	The Materials Reliability Program, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0) final report was issued in December 2008. The mandatory element is that each commercial U.S. PWR unit shall develop and document a PWR reactor internals aging management program (AMP) within thirty-six months following issuance of MRP-227-Rev, 0. With this requirement, the due date the

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
			U.S. PWRs to document their aging management programs will be December 2011.
2010-0005	A-27378	Entergy will supplement the Waterford 3 Cyber Security Plan submittal to clarify the scope of systems described in section 21, scope and purpose, to clarify the Balance-Of-Plant SSCS that will be included in the scope of the Cyber Security Program. Waterford 3 deleted this commitment and is tracking item as soft regulatory obligation while awaiting NRC direction. As per W3F1-2010-0085, based on the recent information and clarification from NERC, Entergy is deferring the proposed supplement to the Waterford 3 cyber security plan until after the NRC's staff has provided direction on how to revise the Cyber Security Plan to reflect the inclusion of Balance Of Plant equipment.	As stated in letter from NSIR to NEI, dated 11/26/1 0, "in light of this Commission policy decision, the NRC staff coordinated with NERC staff to defer licensees' submittals of their revised cyber security plans in accordance with a schedule to be provided by the NRC. The NRC will provide each licensee with regulatory guidance in the form of a template that can be used when supplementing its cyber security plan and a schedule for submittal of the revised cyber security plan. The NRC staff will coordinate with the Nuclear Energy Institute to develop this template in an expeditious manner."
2011-0001	A-27400	Waterford 3 will replace the fibrous insulation on the current Steam Generators with reflective metal insulation. Schedule — Prior to mode 4 after Refueling Outage 17 (RF-17). Waterford 3 changed the commitment to replace the fibrous insulation on the current Steam Generators with reflective metal insulation. Schedule —	Steam Generators will not be replaced during RF-17 due to cladding in bowl area not adhering to base metal; repair activity moving installation of replacement Steam Generators to RF-18. Reflective metal insulation only designed to be attached to replacement Steam Generators. In Staff Requirements Memorandum

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
		Prior to mode 4 after Refueling outage 18	SECY-10-0113 —Closure Options for Generic Safety Issue -191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance, the NRC indicated that resolution of GSI-191 is dependent upon completion of tests and analyses regarding in-vessel effects, which would be needed to allow formulating a path forward by mid-year 2012. As such, the NRC stated that it would be prudent to defer further GSI-191 plant modification actions (such as fibrous material removal) until the tests and analyses are complete.
2011-0002	A-27121	Waterford 3 is in the process of evaluating MRP-146, Materials Reliability Program: management of thermal fatigue in normally stagnant non-isolable Reactor Coolant System branch lines, and these results will be incorporated into the RIS_B Program, if warranted. See W3-ISI-005 request to use ASME Code Case N-716, Committed date: 6/30/2011 Waterford 3 changed the commitment to a committed date of 12/10/13 RF-19 scope freeze milestone).	Waterford 3 is in the process of performing evaluations per MRP-146 guidance. Specifically, the MRP-146S evaluation has not been completed. The initial MRP-146 evaluation was performed and the initial UT examinations of the three susceptible lines were performed in RF-16 with no relevant conditions documented. The new commitment date of RF-19 Scope Freeze Milestone is acceptable because it allows performance of the inspections within the Current ISI 3rd Interval 2nd Period. Performance of these inspections

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
			once per ISI Period would be a conservative inspection schedule considering the results of the initial inspections and the initial evaluation documenting that the lines may screen out of the MRP-146S once it is finalized. RF19 is the last opportunity to perform this inspection within the Current ISI 3 Interval 2nd Period schedule. Selection of the commitment date as RFI 9 Scope Freeze Milestone allows for the inspections to be placed into the original RF-19 scope and allows for the date to reflect changes in the RF-19 start date which are not currently known based on schedule fluxuations and major refuel scope changes made in RF17 and RF18.
2012-0001	P-27254	Compensatory Measure 4 from 2009 NRC Force on Force Exercise. Waterford 3 deleted this commitment	This commitment has been deemed unnecessary due to substantive changes in the Physical Security Plan, compliance with revision to rule (10CFR73.55) change, actions implemented from Root Cause Analysis (CR-WF3-2009-1307) which includes usage of Corrective Action Program in drills and tabletops, and various successful drills and tabletops.

Commitment Change Summary Report

CCEF Number	Commitment Number	Commitment Description	Reason for Change/Deletion
			Additionally, the defensive strategy and tactical deployment procedure has been changed to incorporate a diverse reallocation of resources to ensure adequate protection and capabilities to detect, assess, interdict, and neutralize threats up to and including the design basis threat of radiological sabotage.

10 CFR 50.59 EVALUATION FORM

Sheet 1 of 15

I. OVERVIEW / SIGNATURES[1]

Facility: Waterford-3 **Evaluation # 2010-01/ Rev. #:** 0

Proposed Change / Document: Zink Injection Skid Addition / EC-4019

Description of Change:

To install a skid mounted Zinc Injection System that will accurately deliver a soluble zinc compound, zinc acetate dehydrate, to the Reactor Coolant System (RCS) during normal plant operation. This proposed change will be implemented at Waterford 3 starting in cycle 17.

System Description:

The purpose of the Zinc Injection System is to reduce the shutdown radiation fields, reduce the general corrosion rate of RCS component materials, reduce both the initiation and propagation of primary water stress corrosion cracking (PWSCC) of Alloy 600 and other austenitic stainless steel alloys, and reduction in long-term potential for crud induced power shift (CIPS).

The Zinc Injection System to consist of the zinc injection skid and ½" tubing connecting the skid to the discharge line of the Chemical Addition Pump. The skid will have two parallel pumping trains, each train containing a feed tank, pump, outlet pressure gauge, associated valves & tubing and controls. The individual trains connect to a single discharge point on the skid which ties into CVC-6051 (Zinc Addition Skid Discharge Isolation Valve) located on the discharge side of the Chemical Addition Pump that connects to the suction side piping of the Charging Pumps between the Volume Control Tank and the Chemical Addition Pump. The Charging Pumps then deliver the zinc compound to the RCS.

The Zinc Injection System will be part of the existing Chemical & Volume Control System (CVCS) and the zinc injection skid (CVC-MINJ0001) will be located in the Reactor Auxiliary Building at elevation -4.00 ft (Room B101) near the Chemical Addition Pump. The Zinc Injection System is classified as non-safety, non-seismic and non-quality related system. Existing safety related check valve CVC-606 provides the Chemical and Volume Control system safety class 2 to non-safety piping class break for the line that the non-safety zinc injection skid feeds.

Chemistry procedure CE-002-006 (Maintaining Reactor Coolant Chemistry) is the governing document that will contain the limits, corrective actions, and monitoring program for zinc injection. CE-002-038 (Operation of the Reactor Coolant Zinc Injection Skid) will contain the operating instructions for the skid.

Evaluation Summary:

Background:

The beneficial effects of zinc addition include reduction in shutdown radiation fields, reduction in the general corrosion rate of RCS component materials, reduction in both the initiation and propagation of primary water stress corrosion cracking (PWSCC) of Alloy 600 and other austenitic stainless steel alloys, and reduction in the long-term potential for crud induced power shift (CIPS). These effects result from incorporation of zinc ions into the existing oxide films that have formed on these RCS materials during previous exposure to primary coolant. The RCS chemistry control program at WF3 is modified to allow addition of a soluble zinc compound, zinc acetate dihydrate, to the RCS during normal plant operation.

Evaluations have been performed by Westinghouse to assess the potential impacts on plant operation and to determine the safety significance of adding zinc to the RCS and interconnected systems. These evaluations are

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

listed in the reference section. EPRI Reports also listed in the reference section were reviewed for guidance in establishing the acceptability and impact of zinc injection.

The table below list the important assumptions used in the evaluations. All values listed in the table are based on data or information available in Reference 7.

Table 1
Key Assumptions for Zinc Addition Safety Evaluations

Parameter	Limit
Zinc Concentration, Maximum Target	≤ 40 ppb
Zinc Concentration, Transients	≤ 80 ppb
Maximum Length of Zinc Addition per Cycle	≤ 18 months

The maximum targeted steady-state concentration of zinc in the primary coolant as evaluated by Westinghouse for WF3 is 40 ppb. Plant operation at lower zinc concentrations is bounded by this upper nominal value of 40 ppb. In addition to the limiting target RCS value of 40 ppb, the maximum injection period is 18 months for a given cycle.

Short-term zinc transients are limited to 80 ppb, which is sufficiently below the solubility limit for zinc in the coolant during power operation. Due to the retrograde solubility of zinc compounds with temperature (the solubility of zinc increases with decreasing temperature), the coolant zinc concentration may be observed to increase above this short-term transient value. These increases may occur during shutdown or periods of reduced RCS temperature. Westinghouse will perform an evaluation each cycle outlining the zinc injection strategy using reactor core behavior, primary water chemistry, zinc and corrosion product data, zinc target concentration data, nickel and silica concentration limits. Sustained operation above 40 ppb zinc would require reevaluation. Based on the experience at various operating plants, zinc concentrations are efficiently reduced by the CVCS demineralizers when zinc injection is terminated. Therefore, if high zinc levels are experience due to a temperature reduction, then the zinc concentration would be quickly reduced by letdown purification. Zinc addition will not be resumed until the analyzed zinc concentration falls below the target value of 40 ppb.

WF3 Chemistry procedure CE-002-006 (Maintaining Reactor Coolant Chemistry) is the governing document that will contain the limits, corrective actions and monitoring program for zinc injection.

Materials and Chemistry Effects

Primary System Materials and Components

As a result of the reaction of zinc with existing oxides, the oxide films become thinner, yet more protective. This effect has been demonstrated by laboratory testing at Westinghouse and confirmed by the results of other research organizations.

Laboratory testing indicated that corrosion rates of typical PWR construction materials are reduced by about a factor of at least three or more with approximately 70 ppb-months of zinc exposure. This was achieved by maintaining a zinc concentration of 20 ppb for 3.5 months. These dramatic reductions in corrosion rates were expected to lead to a marked long-term reduction in plant radiation fields.

Since completing the laboratory qualification work, zinc addition has been used at operating PWRs containing Alloy 600 Mill Annealed (MA), Alloy 600 Thermally Treated (IT), and Alloy 690 TT steam generator tube

materials. The Waterford-3 steam generators contain Alloy 600 MA tubing, and the replacement steam generators which will be installed during a future refueling outage will have Alloy 690 TT tubing. No plant has reported a detrimental impact on RCS materials due to zinc addition. This includes the reactor vessel, steam generators, reactor coolant pumps (RCPs), and the CVC system.

Based on laboratory testing and the increasing field experience, there is no evidence to suggest that zinc addition will have an adverse impact on RCS materials, components, or operation. All results support use of zinc addition to reduce corrosion of RCS materials and mitigate PWSCC.

Chemical and Volume Control System

There has been no corrosion test data collected for CVCS materials at CVCS operating conditions, either with or without zinc. Therefore, the extent to which thinning of CVCS oxides by zinc addition occurs, if it occurs at all at the lower operating temperatures of the CVCS, has not been measured. It is Westinghouse's judgment, based on no adverse experience to date, that the presence of ionic zinc in the low-temperature portion of the CVCS may, by the same mechanism that is operative with continuous use of zinc in the RCS, result in lower long-term corrosion rates of CVCS materials (ref. LTR-RCPL-08-43).

EPRI Document 1009568, Overview Report on Zinc Addition in PWRs, reflects zinc injection has not had a detrimental effect on plant equipment (e.g. pump seals, valves, CVCS components) nor resulted in significant increases in radwaste.

The Zinc Injection System will be operated by the W-3 Chemistry Department under CE-002-038. The injection system can be isolated from the CVCS by valve CVC-6051 which is under plant Operations control per OP-002-005. No special provisions for placing in-service or securing the non-safety zinc injection skid are required.

Reactor Coolant Pump (RCP) Seals

The Waterford 3 RCP Flowserve seals are made of tungsten carbide with nickel binder. This seal material is known to leach, particularly in the presence of strong oxidizers. Adding zinc acetate dihydrate to the coolant would not be expected to degrade this material. Although nickel is expected to be released into the coolant from oxide layers of stainless steel and Inconel due to zinc addition, this process is very different from the nickel leaching that has been observed for this RCP seal material. Palisades, which has the same Flowserve seals as Waterford 3, has been adding zinc since 1999 and has had very good seal performance.

Other Primary Components

The potential effects of zinc injection to the following components were also considered in Reference 2.

- Charging pump (CP) components
- Pump mechanical seals associated with the shutdown cooling (SDC) system and safety injection system (SIS) pump seal
- Motor-operated valve and control components
- Pressurizer heater sleeves
- Control element drive mechanisms (CEDMs)
- Heat exchangers in either the CVCS or SDC systems
- RCS instrumentation and their lines
- Valve seat materials, generally Stellite or lower low-wear materials
- Valve stems, with respect to cracking and leakage

Conclusions for Primary Systems Materials and Components

Based on laboratory testing and the increasing field experience, there is no evidence to suggest that zinc addition will have an adverse impact on RCS materials, other primary components or operation. All results support use of zinc addition to reduce corrosion of RCS materials and mitigate PWSCC.

Reactor Coolant pH During Normal Operation

The effect of zinc on reactor coolant pH was evaluated. Based on Farley Unit 2 Cycle 10, with zinc measurements in the range of 22 to 42 ppb, the net effect on the at-temperature pH of the RCS was calculated to vary from 0 to +0.02 pH units. This effect is judged by Westinghouse to be negligible. Therefore, the magnitude of the potential impact of zinc addition on reactor coolant operating pH is small enough to be ignored. Adding zinc to the reactor coolant will have no measurable effect on pH control.

Post-Accident Hydrogen

Metallic zinc will react with acid to generate hydrogen. In the oxidation-reduction reaction, zinc is oxidized from the metallic state to the zinc ion, and the hydrogen ion from the acid is reduced to gaseous hydrogen. For each gram-mole of metallic zinc exposed to acid, one gram-mole of hydrogen is produced. Because zinc only exists in either the 0 or 2 oxidation states, no further hydrogen production is possible once the metallic zinc has been oxidized.

The additive that will be used for zinc addition in Waterford-3, zinc acetate dihydrate, contains zinc as soluble zinc ions. When zinc reacts with existing oxide films, it is either adsorbed on the oxide surfaces as hydrolyzed zinc ions, or it is simply incorporated in the oxide crystal lattice as discrete zinc ions. As a result, if zinc added to the RCS as part of the zinc addition program were to be released to containment following a LOCA, it could only be released as ionic zinc. Any ionic zinc either released from the crystal lattice, desorbed from RCS surfaces, or present in solution in circulating reactor coolant cannot liberate hydrogen, even under a post-LOCA scenario. Therefore, zinc addition cannot contribute to post-LOCA hydrogen generation.

Sump Screen Performance

Westinghouse performed an engineering analysis to show that zinc added to the Waterford-3 RCS will not adversely affect the performance of the emergency core cooling system (ECCS) and containment spray system (CSS).

During active zinc addition, zinc concentration ranges are typically between 5 to 40 ppb (micrograms per kilogram of solution). In a post-LOCA environment, the sump zinc concentration would be significantly more dilute than the RCS concentration due to the introduction of fluids from the safety injection system (SIS). Also, the temperature of the sump fluid would be much less than RCS operating temperature, and zinc compounds are inversely soluble with temperature; as temperature decreases zinc solubility increases. Therefore, the solubility of zinc compounds would increase as the concentration decreased. Under these conditions, precipitation of zinc compounds is unlikely and would have no adverse effects on the performance of the ECCS and CSS.

The analysis confirms that zinc released to the containment sump as a result of zinc addition will not hinder sump screen performance (Reference 3).

Sump pH

Following a loss-of-coolant accident (LOCA), the reactor coolant volume will comprise only a small fraction of the resultant total sump volume. The boric acid concentration of the sump will be high, approximately 2500 ppm boron. As the temperature of the sump decreases, the acid strength of boric acid increases with the decreasing temperature. As a result, trisodium phosphate (TSP) addition is required to raise the pH to a target of 7.0. Since the molar concentrations of boric acid and TSP in the sump are many orders of magnitude greater than the molar concentration of zinc ion, the effect of the zinc ion on pH is negligible.

After multiple cycles of zinc addition, trace concentrations of zinc may be detectable in various boric acid supply tanks Refueling Water Storage Pool (RWSP), mix tanks, accumulators, spent fuel pool, etc.) at Waterford-3.

This occurs in a manner analogous to increasing silica concentrations in these tanks because of recovery, recycling, and reuse of the boric acid. However, unlike silica, the zinc ions are more easily removed by normal resin cleanup practices. Such routine cleanup is commonly performed on the RWST at least once a cycle in anticipation of a refueling shutdown as part of plant shutdown chemistry preparations. Based on the relatively low, perhaps non-detectable, concentrations of zinc in these boric acid sources, their contribution to post-LOCA sump pH is also considered trivial.

Effects on RCS Boron Concentration – Normal Operation

The planned zinc injection solution contains no boron and, therefore, when injected will have a dilution effect on the RCS boron concentration. For most of the fuel cycle, however, the RCS boron concentration may require adjustment to compensate for core burn up. Considering the recommended zinc injection flow rate of approximately 5 ml/min (.00132 gpm), its effect on RCS boron concentration would be insignificant compared to the volume of primary make-up (PMU) water required to lower RCS boron concentrations by 1 ppm. W-3 Chemistry Procedure CE-002-006, Maintaining Reactor Coolant Chemistry, in the Bleed and Feed equation, results in 1620 gallons of PMU is required to reduce the RCS (81,831 gal), boron concentration by 1 ppm at the conservative end of core life when boron concentration are approximately 50 ppm. The zinc injection 5 ml/min flow rate is therefore considered insignificant and will not result in an unexpected RCS boron dilution.

The zinc injection dilution rate is greater than that required for core burn-up for approximately the first 180 days of the cycle. However, the initial use of zinc injection calls for injecting zinc after the first half of the cycle is complete. At boron concentrations associated with operation beyond 180 days, the zinc injection dilution effect is less than that required for core burn-up and will slightly reduce the normal dilution volume added to the RCS by the operator. Should Waterford-3 elect to inject zinc at the start-of-cycle in future cycles, a slight increase in boric acid additions to the RCS may be required to compensate for the dilution from zinc injection. The Waterford-3 Chemistry Dept. monitors boron concentrations and adjusts as necessary.

It is concluded that implementation of a zinc addition program for Waterford-3 does not significantly or unacceptably impact normal RCS boron dilution during the plant operating cycles evaluated.

Radiological Considerations

To support the proposed change to the chemistry control program to employ zinc addition, Westinghouse evaluated (Reference 4) the impact of zinc addition on Waterford-3 primary system radiological conditions. The evaluation conservatively assumes a maximum average primary system zinc concentration of 40 ppb over 18 months of operation. Short-term concentration peaks may occur with a maximum primary system zinc concentration of 80 ppb. The use of either natural or depleted zinc has been addressed.

Various assessments of the radiological aspects of zinc addition for a variety of plants have been performed by Westinghouse. These assessments were based upon the existing experience of the radiological aspects of zinc addition. Evaluations have been required since radioactive Zn-65 is formed by the activation of Zn-64 found in natural zinc. Based on Westinghouse experience, the amounts of Co-58 and Co-60 in the coolant are expected to increase until conditioning of the primary system corrosion product deposits is complete.

The following effects of additional Zn-65 and the additional radiocobalts in the reactor coolant are addressed for implementation and continued usage of zinc addition in Waterford 3:

- the effect of Zn-65 and radiocobalts on the total activity in the CVCS mixed-bed demineralizer,
- the effect of Zn-65 and radiocobalts on the Technical Specifications for the radionuclides in the coolant,
- the impact of Zn-65 and radiocobalts in the coolant on the Equipment Qualification (EQ) dose, and
- other operational considerations.

The pre-zinc baseline coolant measurements of the most recent cycle designs for Waterford 3 were reviewed to determine the effective, long-term, steady-state, full-power values for Co-58 and Co-60. The highest estimated steady state values were found to be 9.16E-05 for Co-58 and 4.19E-6 for Co-60. Conservative multipliers of 61 and 39 for Co-58 and Co-60, respectively, were applied to the Waterford 3 pre-zinc values to obtain the

projected radiocobalt values listed in the Table below.

Table 2
Projected Steady-State Average Coolant Activity Concentrations at 100% Power

Nuclide	Concentration (Note 1)
Zn-65	8.00E-03
Co-58	5.59E-03
Co-60	1.63E-04
Total	1.38E-02

Note 1: Activity concentrations (μCi/ml) are based on pre-zinc values using conservative projections based on the latest available industry zinc data.

These multipliers were based on measurements at other zinc addition plants that used RCS zinc concentrations up to 40 ppb. The value for Zn-65 listed in the Table reflects the maximum observed at any plant performing zinc addition using natural zinc; the contribution from depleted zinc would be negligible. Elevation of the RCS radio-cobalt activities indicates that zinc is having its desired effect on plant surfaces, resulting in reduced dose rates from these surfaces. The elevation in the radiocobalt activity is expected to be temporary, lasting only until the plant oxide layers are fully conditioned, at which time most of the dose rate reduction will be achieved. The Zn-65 activity should be anticipated to be present in the RCS coolant if, and as long as, natural zinc is used at Waterford-3.

The above values are used in the following assessment of the impact of zinc addition planned for Waterford-3, using either natural or depleted zinc. Since the Zn-65 value is based on the use of natural zinc acetate, the values above are conservative if depleted zinc is actually used, since there would effectively be no contribution from Zn-65.

The radionuclides impacted by zinc addition consist only of Co-58, Co-60 and Zn-65. Review performed on UFSAR Chapter 11 (Radioactive Waste Management) Tables 11.1-2, 11.1-3, 11.1-9, 11.1-10, 11.1-11, 11.2-13, 11.4-7 & 11.4-8 and Chapter 12 (Radiation Protection) Tables 12.2-4, 12.2-5, 12.2-7, 12.2-8, 12.2-10, 12.2-11, 12.2-12, 12.2-15a, 12.4-1a & 12.4-4 showed that the concentrations of these radionuclides are bounded by the UFSAR Table 11.1-3 (Average Reactor Coolant Radioisotope Concentration – No Gas Stripping). Each of these values for Waterford 3 is bounded by predictions generated for other plants. Therefore, no changes are required to any of the Tables mentioned above (Reference 10).

Effect on Demineralizer Activity

Based on the conservative, post-zinc activity concentrations listed above, demineralizer activity concentrations for Co-58, Co-60, and Zn-65 could reach 1.23E+02 μCi/cc, 1.87E+01 μCi/cc, and 4.93E+02 μCi/cc, respectively. Each of these values for Waterford-3 is bounded by predictions generated for other plants. These estimates include conservative assumptions related to 18-month continuous operation at 40 ppb and no resin replacement.

For operational conditions in which the demineralizer operating characteristics (letdown flow rate, resin replacement frequency, etc.) remain the same, increases in dose rates will be observed. This increase is not expected to have an impact on operation of the plant since the mixed-bed source will build up in a gradual manner over the cycle operation period. Radiation monitors and other health physics surveys can be utilized to monitor this expected increase. To date, no plants operating with zinc injection have reported problems with increased demineralizer dose rates.

Based on the foregoing, operation with zinc addition is not expected to significantly impact the occupational

radiation exposure received by plant personnel.

Effect on Technical Specifications

Per Waterford 3 Technical Specifications (Reference 9), there are no limits for individual radionuclides in the reactor coolant. The only limit is given in the terms of the sum of the average beta and gamma energies. Isotopes other than iodines, with half lives greater than 15 minutes are considered. This term is called Ē, or the average disintegration energy. The average value is limited to 100/ Ē µCi/g. The typical design value of Ē for all nuclides considered is about 0.31 (Reference 10), thus giving the coolant a mean value of 323 µCi/g (= 100 µCi/g/ 0.31).

From Table 2, the maximum coolant activity anticipated due to zinc addition is 1.38E-02 µCi/ml. This value is in the order of magnitude lower than the 323 µCi/g derived from the average disintegration energy considerations. Therefore, the addition of zinc at Waterford 3 will not have a significant impact on the coolant concentration limits, so the limit on specific activity in Technical Specification 3/4.4.7 will not require modification due to zinc addition.

Effect on Equipment Qualification Dose

Section 12.3A of the Waterford-3 UFSAR identifies the basis for the environmental qualification standards at Waterford-3. The key conservative assumption (with regard to nuclide contribution) is use of the core melt fission product release of 100% noble gases, 50% of halogens, and 1% of the remaining isotope activity. The total contribution from Co-58, Co-60, and Zn-65 (1.38E-02 µCi/ml) due to zinc injection is insignificant compared to that of the fission products assumed above.

Effect on Fuel

An evaluation of the impact of zinc addition on fuel and core behavior was performed for Waterford 3 by Westinghouse and found that fuel performance criteria are unaffected by zinc addition (Reference 5). These criteria include clad corrosion, rod internal pressure, cladding stress, cladding strain, cladding fatigue, clad collapse, fuel centerline melt, end plug weld integrity and fuel rod growth. Zinc addition has been evaluated as acceptable with respect to effect on core nuclear or thermal / hydraulic characteristics

Effect on Non-LOCA Transient Analyses

An evaluation of the impact of zinc addition on the non-LOCA safety analyses was performed by Westinghouse. The non-LOCA safety analyses are not violated by either unborated water addition or by zinc from the Zinc Injection system. This includes the Uncontrolled Boron Dilution event for all six operational modes. Since there is no increase in the CVC system charging pump flow in all six operational modes, there is no increase in the boron dilution rate and no change in the time to criticality. Hence, from a non-LOCA transient analysis standpoint, the Zinc Injection system can be used in all six operational modes since there is no impact on the non-LOCA safety analyses. Therefore, the current non-LOCA transient analysis results remain valid, and no acceptance criteria are violated. In addition, there are no changes to the non-LOCA transient analysis inputs to the core operating limit supervisory system (COLSS) and core protection calculator system (CPCS) thermal margin requirements.

Effect on Non-LOCA Transient Dose Consequences

Westinghouse evaluation shows that zinc injection potentially impacts the levels of the following isotopes in the RCS and the secondary system: Zinc-65, Cobalt-58 and Cobalt-60. Although the evaluation was performed for a Westinghouse Fleet plant, specifically Surry Units I and 2, it is similar to the evaluation necessary for Waterford-3, a CE-fleet plant, because the Surry plants and Waterford-3 are PWRs. A similar analysis was performed to apply Callaway (a Westinghouse Fleet plant) analysis methods and results to Calvert Cliffs, a CE-

fleet plant.

Waterford-3 exclusion area boundary (EAB), low population zone (LPZ) and control room (CR) radiological dose calculations for non-LOCA safety analysis design basis events (DBEs) are based on the alternate source term (AST) analysis method described in USNRC Regulatory Guide 1.183. Table 5 of Regulatory Guide 1.183 identifies the radionuclide groups (i.e. the isotopes of the various radioactive elements) that are considered when computing AST radiological doses. Neither zinc nor cobalt is included in this table. Hence, changes in Zn-65, Co-58 and Co-60 levels due to zinc injection will not impact any of the radiological dose calculations currently presented in Chapter 15 of the Waterford-3 UFSAR. Nonetheless, an evaluation of the potential impact of changes in the Zn-65, Co-58 and Co-60 levels on the EAB, LPZ and CR radiological doses was performed.

It is assumed that zinc injection could increase Zn-65 concentration in the RCS to a value as high as 8.0E-03 µCi/g. Unlike the halogen and noble gas activities that typically dominate a radiological dose calculation Zn-65 is not volatile and would not become airborne. The assumed 8.0E-03 µCi/g is sufficiently low to be insignificant when compared with fission products such as the radioactive iodine (halogen) isotopes which are assumed to be present at an equilibrium concentration of 1.0 µCi/g dose equivalent I-131. Thus, the Zn-65 concentration has negligible impact on the radiological consequences of non-LOCA safety analysis EAB, LPZ and CR radiological doses presented in Chapter 15 of the Waterford-3 UFSAR.

The evaluation revealed that expected cobalt isotope concentrations are very similar to the maximum concentrations determined for Surry Units 1 and 2 following implementation of zinc injection. The change in isotope concentrations did not have a significant impact on the safety-related and plant operational radiological considerations at Surry Units 1 and 2. It is, therefore, expected that changes in the Co-58 and Co-60 concentrations at Waterford-3 will have negligible impact on the radiological consequences of the non-LOCA safety analysis EAB, LPZ and CR radiological doses presented in Chapter 15 of the UFSAR.

Technical Specification Review

The discussion above of the radiological effects of zinc injection in the section "Radiological Considerations" confirmed that no revision to Technical Specifications 3/4.4.7, Specific Activity, is needed. Technical Specification 3/4.4.6, Chemistry, contains limits on dissolved oxygen, chloride and fluoride. As described in the Bases B3/4.4.6, these limits ensure that RCS corrosion is minimized and they reduce the potential for RCS leakage or failure due to stress corrosion. Limits on zinc are not needed for these purposes. Effects of zinc addition include reduction in the general corrosion rate of RCS component materials and reduction in both the initiation and propagation of PWSCC of Alloy 600 and other austenitic stainless steel alloys (Reference 2). As a result, no revision to Technical Specification 3/4.4.6 is needed due to zinc injection.

Review of the Technical Specifications confirmed that no other changes to other Technical Specifications are required.

Effect on Plant Procedures

Zinc addition will affect details of the procedures for CVCS operation, chemistry control, chemistry sampling and plant start-up and shutdown evolutions. The changes involve accounting for zinc injection flow and adjusting zinc concentration. They do not, however, adversely affect operation of any SSC that are important to safety as described in the UFSAR. Zinc addition does not affect emergency procedures. The complexity of chemistry sampling, boration and dilution evolutions is not increased.

Chemistry procedure CE-002-006 (Maintaining Reactor Coolant Chemistry) is the governing document that will contain the limits, corrective actions and monitoring program for zinc injection into the RCS during normal plant operation.

Zinc Injection System Equipment (Skid)

The skid is supplied by Radiological Solutions, Inc. (RSI). The skid has two trains supplying the main header. Each train has a tank (5 gallon capacity), metering pump, isolation valves, relief valve and check valve. The main header of the skid connects directly to valve CVC-6051 (Zinc Addition Skid Discharge Isolation Valve).

The skid was designed with the necessary valves and tubing to allow for operation of either metering pump with either tank (cross-connected). However, the metering pumps cannot be operated simultaneously. A control panel is provided for pump selection, feed rate adjustment for the pump in service, and local indication of the pump in service.

The skid will operate on 60Hz, single phase, 120 VAC power with a total system electrical demand of less than 5 amps and will be installed in the Reactor Auxiliary Building, elevation -4.0 ft. (Room B101).

Chemistry procedure CE-002-038 (Operation of the Reactor Coolant Zinc Injection Skid) will contain the operating instructions for the skid.

References:

1. LTR-RCPL-08-43, "10CFR50.59 Applicability Determination and Screen for Waterford 3 Zinc Injection (EVAL-08-112)", August 19, 2008 by Westinghouse

2. LTR-CDME-08-41, "Chemistry and Materials Inputs for Waterford Unit 3 Zinc Addition 10CFR50.59 Safety Evaluation", August 18, 2008 by Westinghouse

3. LTR-CDME-08-40, "Impact of Zinc Addition on Waterford 3 Containment Sump Screen Performance Under Post-LOCA Conditions", August 11, 2008 by Westinghouse
4. CN-REA-08-18, "Radiation Analysis Evaluation for Zinc Addition at Waterford Unit 3", August 19, 2008 by Westinghouse

5. CN-WTR316-019, Rev. 1, "Waterford 3 Cycle 16 Fuel Rod Corrosion and Zinc Injection Calculation", August 5, 2008 by Westinghouse

6. CN-CDME-07-26, "Waterford 3 Zinc Injection Effect on boron Dilution", February 6, 2008 by Westinghouse

7. CWTR3-08-385, "Receipt of Entergy's Letter dated July 10, 2008 Documenting Entergy's Review and Confirmation of Inputs and Assumptions for the Waterford 3 Zinc Injection Addition Program", July 22, 2008 by Westinghouse

8. Waterford 3 USAR Chapter 9, section 9.3.4 (Chemical & Volume Control System), Chapter 11 (Radioactive Waste Management), Chapter 12 (Radiation Protection), Chapter 15, section 15.4.1.5 (CVCS Malfunction – Inadvertent Boron Dilution).

9. Waterford 3 Technical Specifications through Amendment 213.

10. Westinghouse Radiation Analysis Manual – Standard 412, Revision 3, November 1978.

11. EPRI Report 1009568, "Overview Report on Zinc Addition in Pressurized Water Reactors – 2004"

12. EPRI Report 1013420, "Pressurized Water Reactor Primary Water Zinc Application Guidelines"

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

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

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

Preparer: Alex Abella / Electronic / DP Engineering / 09-15-09
Name (print) / Signature / Company / Department / Date

Reviewer: T. Hempel / IAS Electronic / Entergy / Design Engineering / 01-13-10
Name (print) / Signature / Company / Department / Date

OSRC: Ken Christian / IAS Electronic / Entergy / Nuclear Safety / 02-25-10
Chairman's Name (print) / Signature / Date

W310-02
OSRC Meeting #

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II. 50.59 EVALUATION

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

Does the proposed Change:

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

   BASIS:

   The following UFSAR Sections were reviewed:
   - 15.4.1.5, CVCS Malfunction (Inadvertent Boron Dilution)
   - 15.5.1.1, Chemical & Volume Control System Malfunction
   - 15.5.2.1, Chemical & Volume Control System Malfunction with a Concurrent Single Failure of an Active Component

   The review found that the addition of Zinc Injection System does not change or alter the accident analysis previously evaluated in the UFSAR.

   The addition of zinc into the RCS water chemistry does not introduce the possibility of a change in the frequency of an accident because the addition of zinc into the primary water chemistry is not an accident initiator, and no new failure modes are introduced. The zinc injection flow path does not alter the CVCS design function and operation. The zinc injection system only operates when the plant is in operation.

   The effect of zinc acetate dihydrate on the wetted surfaces in the RCS and associated equipment does not introduce a mechanism to accelerate degradation of equipment or components. It is generally accepted that zinc is effective at mitigating the extent and consequences of general corrosion of both austenitic stainless steels and nickel-base alloys in primary water. Considering the 5 ml/hr zinc injection flow rate, measurable RCS boron dilution or reactivity changes effects should not result.

   The Zinc Injection System does not affect the overall system performance or reliability in a manner that could change the likelihood of accident occurring in the RCS and CVCS. The skid operates independent from the plant piping system. It has its own instrumentation/controls to regulate the zinc injection. If a problem is encountered by the skid, valve CVC-6051 can be closed and the zinc injection system can be isolated and the problem fixed without affecting the CVCS. Chemistry procedure CE-002-006 (Maintaining Reactor Coolant Chemistry) will contain the limits, corrective actions and monitoring program for zinc injection. CE-002-038 (Operation of the Reactor Coolant Zinc Injection Skid) will contain the operating instructions for the skid.

   The proposed change meets the applicable standards of the CVCS and does not affect overall system performance in a manner that could have an accident consequence therefore, the installation of the Zinc Injection System will not increase the frequency of occurrence of an accident previously evaluated in the UFSAR.

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

BASIS:

Westinghouse evaluations and industry operating experience have shown no detrimental effects on systems, structures and components (SSC) exposed to reactor coolant containing zinc at the concentrations that will be implemented. The low rate of zinc injection flow has a negligible effect on CVCS control of RCS boron concentration and the rate at which RCS boron concentration could be inadvertently reduced. Zinc injection does not affect how the design functions and operations of the SSC are accomplished or controlled.

The zinc injection skid meets the original design specifications for materials and construction of the CVCS. The Zinc Injection System is classified as non-safety, non-seismic and non-quality related system.

It is established by laboratory testing at Westinghouse and extensive industry experience that zinc addition is not an accident initiator and does not increase a likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

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

BASIS:

Zinc addition provides benefits in reducing the shutdown radiation fields, reduction in the general corrosion rate of RCS component materials, reduction in both the initiation and propagation of primary water stress corrosion cracking (PWSCC) of Alloy 600 and other austenitic stainless steel alloys, and reduction in the long-term potential for crud induced power shift (CIPS). These effects result from incorporation of zinc ions into the existing oxide films that have formed on these RCS materials during previous exposure to primary coolant. The RCS chemistry control program at WF3 is modified to allow addition of a soluble zinc compound, zinc acetate dihydrate, to the RCS during normal plant operation.

An evaluation of the impact of zinc addition on fuel and core behavior found that fuel performance criteria are unaffected by zinc addition. These criteria include clad corrosion, rod internal pressure, cladding stress, cladding strain, cladding fatigue, clad collapse, fuel centerline melt, end plug weld integrity and fuel rod growth. Zinc addition has been evaluated as acceptable with respect to effect on core nuclear or thermal / hydraulic characteristics. The implementation of zinc addition to the RCS does not affect the design basis limits for fission product barriers namely: Fuel Cladding, RCS Boundary and Containment. This proposed change has no adverse radiological effect.

To support the proposed change to the chemistry control program to employ zinc addition, Westinghouse has evaluated the impact of zinc addition on WF3 primary system radiological conditions. This evaluation conservatively assumes a maximum average primary system zinc concentration of 40 ppb over 18 months of operation. Short-term concentration peaks may occur with a maximum primary system zinc concentration of 80 ppb. Review of UFSAR section 15.4.1.5, CVCS Malfunction (Inadvertent Boron Dilution) found that zinc injection will not result in an increase in the consequences of an accident previously evaluated in the UFSAR.

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

BASIS:

The Zinc Injection System is independent from the existing piping system and is classified as non-safety related. The tie-in point for the skid is valve CVC-6051 and the position of this valve will be controlled by Operations. CVC-6051 is a ½" SS globe valve and is classified as non-safety related. The injection skid and components will be controlled under Chemistry procedure CE-002-038. Failure of these components does not have any consequences that would result in radiological release. The zinc injection limits and monitoring will be controlled under Chemistry procedure CE-002-006 (Maintaining Reactor Coolant Chemistry).

Westinghouse evaluation 08-112 reflects a zinc concentration maximum target of <40 ppb, and a zinc concentration from transients at 80 ppb. These values provide sufficient margin when considered with the approximate 5 ppb zinc addition value currently recommended. Chemistry procedure CE-002-006 will control the zinc injection rates based on the Westinghouse recommendations post chemistry and core behavior review which is required prior to commencing injection each plant cycle. If zinc concentrations are found above the prescribed values, injection will cease until the CVC ion exchanges reduce the zinc concentration to the target level. Nuclear plants currently implementing a zinc addition strategy have on occasion exceeded the target concentration due to overfeeding without noticeable consequence.

The addition of zinc into the RCS does not introduce a change in the consequences of a malfunction of an SSC important to safety because zinc addition will not cause pumps, valves or heat exchangers to malfunction and result in a radiological release to the environment. Zinc addition has no effect on SSC used to mitigate the consequences of postulated accidents because it is not an accident initiator. Therefore, implementation of zinc addition does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

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

BASIS:

Review of the accident analysis in UFSAR Chapter 15 for the CVCS determines that these types of accidents are not affected by the implementation of the Zinc Injection System. Since the operation of the zinc injection skid is independent from the CVCS, any malfunction of the skid will have no effect on the CVCS and RCS. The skid can be easily isolated by closing valve CVC-6051 per procedure CE-002-006. The installation and operation of the Zinc Injection System will be governed by approved procedures to minimize any risk associated with new equipments.

The addition of zinc in the RCS does not introduce the possibility of a new accident because the zinc addition is not an initiator of any accident and it does not introduce a new failure mode in systems that provide fission product barriers and mitigate postulated accidents. Zinc addition does not increase the initiation or progression of corrosion in primary pressure boundary materials. Any postulated effect of zinc addition on the integrity of the pressure boundary would be bounded by the previously evaluated pipe breaks. The postulated addition of water by the zinc injection skid into the RCS is bounded by existing boron dilution evaluations. Therefore, zinc addition does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

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

BASIS:
The zinc injection skid satisfies the design requirements of the CVCS. The zinc travels on the same flow path as the existing CVCS piping and thus, no new type of failure mode is introduced to any systems or components. Malfunctions in the zinc injection skid, including loose parts and foreign objects from the skid, will not change the failure modes of any systems and components.

Since the zinc injection skid uses the CVCS to introduce zinc into the RCS via the charging pumps, the following malfunction were reviewed:

1. CVCS Malfunction (inadvertent Boron Dilution) (USAR Section 15.4.1.5)
2. Chemical &Volume Control System Malfunction (USAR Section 15.5.1.1)
3. Chemical &Volume Control System Malfunction with a Concurrent Single Failure of an Active Component. (USAR Section 15.5.2.1)

The review determined that the impact of the introduction of zinc into the CVCS would not change the results of the three malfunctions listed above that have been previously evaluated in the UFSAR.

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

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

BASIS:

The implementation of zinc addition to the RCS does not affect the design basis limits for the three fission product barriers namely: Fuel Cladding, RCS Boundary and Containment.

An evaluation of the impact of zinc addition on fuel and core behavior found that fuel performance criteria are unaffected by zinc addition. These criteria include clad corrosion, rod internal pressure, cladding stress, cladding strain, cladding fatigue, clad collapse, fuel centerline melt, end plug weld integrity and fuel rod growth. Zinc addition has been evaluated as acceptable with respect to effect on core nuclear or thermal / hydraulic characteristics.

The zinc injection skid conforms to the design requirements of the CVCS and the reactor coolant pressure boundary will continue to satisfy the stress and fatigue limits of the ASME code. The design and operating pressure of the skid is bounded by the design criteria of the CVCS and will not affect the Containment design pressure.

Therefore, zinc addition does not result in a design basis limit for fission product barrier as described in the UFSAR being exceeded or altered.

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

BASIS:

No method of evaluation is affected by the installation of the Zinc Injection System. The methods used to evaluate the response of the plant to postulated accident conditions are not changed by the addition of zinc to the RCS. Zinc addition does not require any changes to the methods used to evaluate radioactive dose rate. The methods used to evaluate the reactor coolant pressure boundary integrity are not changed. Therefore, the addition of zinc into the RCS does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

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

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I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # / Rev. #:** 2010-06 / R-0

Proposed Change / Document: EC 14765, SI-405A(B) Bypass Fill / Equalization Line Addition

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

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

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

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

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

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

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

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

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

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Preparer: Thomas R. Hempel / IAS electronic / Entergy / Design Engineering / 03-30-10
Name (print) / Signature / Company / Department / Date

Reviewer: William J. Steelman / IAS electronic / Entergy / Licensing / 04-07-10
Name (print) / Signature / Company / Department / Date

OSRC: Keith Nichols / IAS Electronic / Entergy / Engineering Director/ / 05-05-10
Chairman's Name (print) / Signature / Date

W3 10-08
OSRC Meeting #

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II. 50.59 EVALUATION

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

Does the proposed Change:

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

BASIS: The relevant accident for this modification is a loss of coolant accident (LOCA). The addition of the fill bypass line decreases the likelihood of a pressure transient occurring when SI-405A(B) is opened. During operation, the new solenoid valve in the bypass line will be closed and will have power removed by opening the breaker for the SI-405A(B) control circuit. The control circuit has been designed to eliminate the likelihood of an intersystem LOCA through use of the same RCS Pressure interlock for both parallel valves. This modification adds about 10 feet of ¾" piping to each train of the shutdown cooling suction piping inside containment. Because the new piping and valves are designed, analyzed and qualified to the same ASME Section III Class 1 and 2 requirements as the existing piping and valves, this change does not result in more than a minimal increase in the frequency of occurrence of a LOCA. In addition, the valves are designed and qualified to meet containment isolation requirements.

The replacement combined control switch CS-2 on the remote shutdown panel (RSP) is used to control both valves only during a main control room evacuation scenario. These valves cannot be operated from this location as long as the RCS pressure interlock is functioning and power is removed at the DC distribution panel. The interface pressure boundary also is part of the SI-401 valve function, and these valves are in series to provide backup to each other. Since these valve interlocks remain functional and the use of these valves is not until cold shutdown, the combined control switch function does not result in a more than minimal frequency of occurrence of an accident previously evaluated in the FSAR.

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

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

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

FSAR Section 15.6, Decrease in reactor coolant system inventory

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

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

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

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

transients.

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

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

The replacement combined control switch CS-2 on the remote shutdown panel (RSP) is combined because of space limitations on the panel elevation such that the controls remain in the same functional location for these valves and their interface function with the related valves SI-401 and SI-407. To facilitate the different valves operating from the same switch, opening the fill valve has been designed to require the operator to push in the switch knob instead of turning the switch knob as is normally done. This allows the turning of the knob to be retained for the main valve SI-405, the same as the present switch is operated for the present valve. The push function to open is engraved on the indication position for open of the fill valve SI-4052, which is the same location as the indication for this valve on the Main Control Room switch. Since the switch has only a push function, and no pull function is available, the standard use of pull to open cannot be used. Also, this limits the use of knob in-out to one function for the new valve. Closure of the fill valve is not necessary after the main valve is opened, except for isolation of the SDC line when both valves are required to be closed, so closure of the fill valve is part of closure of the main valve by the operator. The use of this switch is only during an evacuation of the MCR during a fire, where procedures will be closely followed, and is coupled with the fill valve indication clearly engraved with the push instruction. Additionally, the RCS pressure interlock remains in service when the valves are controlled from this RSP location. Therefore, this replacement switch with dual valve control function is not more than minimally adverse to the frequency of a malfunction previously evaluated in the FSAR.

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

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

BASIS: The change proposed adds approximately 10 feet of ¾" diameter piping inside the containment building and is designed to meet reactor coolant pressure boundary requirements. The consequences of any failure of this piping, or the valves in the piping, are bounded by existing analyses in the UFSAR for both large break and small break LOCAs. The new solenoid valve SI-4052A(B) will function in parallel with SI-405A(B) to provide a redundant high/low pressure interface and intersystem LOCA boundary with upstream valve SI-401A(B). The new piping and valves also meet the containment isolation / penetration requirements.

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

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

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

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

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

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

BASIS: The types of accidents that could be caused by this modification are loss of coolant (due to piping failure) or containment boundary leakage (due to valve failure). These present UFSAR accident evaluations are considered not impacted by the change proposed.

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

No new accident initiator or types are created by this change as the new piping system will meet the necessary code and design requirements.

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

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

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

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

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

BASIS: All analyses performed to support this modification are consistent with existing UFSAR described methodologies. Considering the proposed change is limited in scope to the Reactor Coolant / Shutdown Cooling System, the previously approved methods used to evaluate reactor coolant pressure boundary were used without change.

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

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I. OVERVIEW / SIGNATURES[1]

Facility: WF3

Evaluation # / Rev. #: 0
2010-12

Proposed Change / Document: EC-18232

Description of Change: This temporary modification will install a temporary control switch in the control circuit for the Fuel Pool Pumps A & B to bypass the spent fuel pool low low level pump trip during RF16 while SUPS AB bus is de-energized. SUPS AB will be de-energized during RF16 to support Bypass Switch replacement. SUPS AB provides control power to the control relay RLL for the fuel pool pumps A & B control circuits. It is estimated that it will require about 2 hours to de-energize SUPS AB and connect a temporary power source to re-energize SUPS AB bus while the SUPS AB Bypass Switch is replaced. After SUPS AB Bypass Switch is replaced, SUPS AB bus has to be de-energized again to restore it to its normal configuration. This again will remove control power to the control relay for about 2 hours. A temporary control switch will be installed to bypass the low-low level pump trip only during the time that the SUPS AB bus is de-energized for changing power sources. The high and low level alarms for the spent fuel pool will also be disabled during the time that the SUPS AB bus is de-energized. A contingency monitoring plan will be put in place to continuously monitor spent fuel pool level and contact Operations if any deviations are identified during the time the SUPS AB bus is de-energized. This temporary modification installs a temporary control switch bypassing the RLL contacts allowing the fuel pool pumps A & B to operate normally during the time the SUPS AB bus is de-energized to ensure continuity of spent fuel pool cooling during the time SUPS AB bus is de-energized. WO# 212017 task 1 will install this temporary modification and WO# 212017 task 2 will restore the circuit to its normal configuration.

This temporary modification disables a function specifically described in the FSAR which is to trip the Fuel Pool Pumps automatically on low Spent Fuel Pool level. This temporary modification credits manual actions to monitor Spent Fuel Pool water level and trip the Fuel Pool Pumps in the event of low level. Crediting a manual function for an automatic function is considered an adverse affect that requires performance of a safety evaluation under 10CFR50.59. This safety evaluation is being performed now as it was identified that the PAD that was originally performed to support this temporary modification was not adequate.

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

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

N/A

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

Preparer: W. E. Day / WEDay / EOI / SE-EFIN / 12-8-10
Name (print) / Signature / Company / Department / Date

Reviewer: R. Tran / RTran / EOI / DE / 12-8-10
Name (print) / Signature / Company / Department / Date

OSRC: Keith Richdy Kirk Lich 12/9/10
Chairman's Name (print) / Signature / Date
OSRC Meeting # 10-18

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

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II. 50.59 EVALUATION

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

Does the proposed Change:

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

   BASIS: THERE IS NO ACCIDENT PREVIOUSLY EVALUATED IN THE FSAR THAT COULD BE CAUSED AS A RESULT OF BYPASSING THE SPENT FUEL POOL PUMPS LOW LEVEL PUMP TRIP. FUEL POOL LEVEL WILL BE MONITORED BY AN OBSERVER STATIONED ON THE FUEL HANDLING BUILDING REFUELING FLOOR. SHOULD A DROPPING FUEL POOL LEVEL BE OBSERVED, OPERATIONS WILL BE NOTIFIED.

   THIS CHANGE DOES NOT INITIATE OR AFFECT THE FREQUENCY OF ANY ACCIDENT. THE ONLY ACCIDENTS THAT ARE EVALUATED IN THE UFSAR ASSOCIATED WITH THE FUEL POOL COOLING SYSTEM ARE A FUEL HANDLING ACCIDENT AND A FUEL POOL COOLING SYSTEM BREAK. EACH OF THESE ACCIDENTS ARE INDEPENDENTLY EVALUATED AND THE FREQUENCY OF OCCURRENCE OF EACH OF THESE EVENTS CANNOT BE AFFECTED BY FAILURE OF THE FUEL POOL COOLING PUMPS TO TRIP ON LOW FUEL POOL LEVEL AS NO CREDIT IS TAKEN FOR THE OPERATION OF THESE PUMPS IN EITHER ACCIDENT SCENARIO.

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

   BASIS: ENSURING THE FUEL POOL PUMPS REMAIN IN SERVICE IS NECESSARY TO REMOVE DECAY HEAT DURING RF16 WHILE THE FULL CORE IS OFF LOADED INTO THE SPENT FUEL POOL AND MAINTAIN FUEL POOL TEMPERATURE AT OR BELOW 155 °F.

   INSTALLING A JUMPER ACROSS THE LOW LEVEL PUMP TRIP CONTACTS ALLOWS THE FUEL POOL PUMPS TO REMAIN IN SERVICE DURING THE SHORT PERIOD OF TIME POWER IS REMOVED TO THE CONTROL POWER CIRCUIT THAT FEEDS THE FUEL POOL LEVEL SWITCH (~2 HOURS WHILE REMOVING AND RESTORING CONTROL POWER TO SUPPORT MAINTENANCE ON EACH OF TWO INSTANCES). DURING THIS TIME THE FUEL POOL PUMPS WILL NOT TRIP AUTOMATICALLY ON LOW LEVEL AND ALARMS WILL NOT BE GENERATED TO NOTIFY OPERATORS OF A LOW LEVEL CONDITION IN THE SPENT FUEL POOL. THE PURPOSE OF THE AUTOMATIC LOW-LOW LEVEL TRIP LOGIC FOR THE FUEL POOL PUMPS IS TO PROTECT THE PUMPS FROM DAMAGE SHOULD THE LEVEL DROP BELOW THE MINIMUM LEVEL REQUIRED TO MAINTAIN MINIMUM NET POSITIVE SUCTION HEAD. THIS ENSURES THAT THEY WILL BE AVAILABLE WHEN THE LEVEL IS RESTORED TO NORMAL. ADMINISTRATIVE CONTROLS WILL BE ESTABLISHED TO CONTINUOUSLY MONITOR THE FUEL POOL LEVEL AND NOTIFY OPERATIONS SHOULD FUEL POOL LEVEL LOWER. THIS ADMINISTRATIVE CONTROL ENSURES THAT THE LIKELIHOOD OF DAMAGE TO THE FUEL POOL PUMPS DUE TO LOW LEVEL IS NOT INCREASED. WITH DROPPING FUEL POOL LEVEL, OPERATIONS WOULD ENTER OP-901-513, SPENT FUEL POOL COOLING MALFUNCTION, AND ATTEMPT TO MAINTAIN FUEL POOL LEVEL BY ADDING MAKEUP FROM THE REFUELING WATER STORAGE POOL OR THE CONDENSATE STORAGE POOL. ENTERING OFF NORMAL PROCEDURE OP-901-513 IS THE RESPONSE THAT OPERATIONS WOULD IMPLEMENT FOR SPENT FUEL POOL LOW LEVEL WHETHER OR NOT TEMPORARY MODIFICATION EC-18232 IS INSTALLED AND THE TEMPORARY MODIFICATION WOULD NOT DIMINISH THE EFFECTIVENESS OF THE OFF NORMAL RESPONSE.

   SHOULD A FUEL HANDLING ACCIDENT OCCUR WHILE THE AB BUS IS DEENERGIZED, PERSONNEL NOT INVOLVED IN FUEL HANDLING OPERATIONS (I.E., THE POOL LEVEL MONITOR) WOULD BE EVACUATED FROM THE FUEL HANDLING BUILDING. HOWEVER, IT IS NOT CREDIBLE THAT A FUEL HANDLING ACCIDENT WILL OCCUR AT THE SAME TIME AS A PIPE BREAK DURING THE SAME SHORT TIME FRAME THAT THE TEMPORARY MODIFICATION IS INSTALLED. TWO DESIGN BASIS ACCIDENTS, FUEL HANDLING ACCIDENT AND A PIPE BREAK THAT LOWERS SPENT FUEL POOL INVENTORY, ARE NOT POSTULATED TO OCCUR AT THE SAME TIME. THUS, THE MONITOR WILL REMAIN AVAILABLE TO NOTIFY THE CONTROL ROOM TO TRIP THE PUMPS AND PREVENT THE MALFUNCTION OF THE PUMPS ON A LOWERING LEVEL.

LOCAL MONITORING OF SPENT FUEL POOL LEVEL IS A COMPENSATORY ACTION IN LIEU OF AUTOMATICALLY TRIPPING THE FUEL POOL PUMPS ON LOW SPENT FUEL POOL LEVEL. INFORMATION NOTICE IN 97-78 HAS NINE SPECIFIC ITEMS AS DISCUSSED BELOW TO CREDIT MANUAL ACTIONS IN PLACE OF AN AUTOMATIC ACTION:

1. SPECIFIC OPERATOR ACTIONS REQUIRED - WORK ORDER WO-198950 PROVIDES SPECIFIC INSTRUCTIONS TO STATION A FUEL POOL LEVEL MONITOR TO INFORM OPERATIONS SHOULD A FUEL POOL LOW LOW LEVEL OCCUR SO THAT APPROPRIATE ACTIONS ARE TAKEN INCLUDING MANUAL TRIPPING OF THE FUEL POOL PUMPS A & B.

2. POTENTIALLY HARSH OR INHOSPITABLE ENVIRONMENTAL CONDITIONS EXPECTED - THERE ARE NO HARSH OR INHOSPITABLE ENVIRONMENTAL CONDITIONS EXPECTED. THE FUEL POOL LEVEL MONITOR WILL BE STATIONED IN THE FUEL HANDLING BUILDING ON THE REFUELING FLOOR WHICH IS DESIGNED FOR HABITATION BY PLANT PERSONNEL. SHOULD LEVEL DROP, IT CANNOT DROP BELOW +40 FT 6 IN DUE TO SIPHON BREAKERS IN PIPING TO AND FROM THE SPENT FUEL POOL, SO THERE IS AN INSIGNIFICANT RISE IN DOSE LEVELS THAT COULD OCCUR WHICH WOULD NOT RESULT IN AN INHOSPITABLE ENVIRONMENT FOR THE FUEL POOL LEVEL MONITOR.

3. GENERAL DISCUSSION OF THE INGRESS/EGRESS PATHS TAKEN BY OPERATORS TO ACCOMPLISH FUNCTIONS – NORMAL CONTROLS FOR ENTRY AND EXIT FROM THE FUEL HANDLING BUILDING ARE SUFFICIENT TO STATION A FUEL POOL LEVEL MONITOR IN THE FUEL HANDLING BUILDING.

4. PROCEDURAL GUIDANCE FOR REQUIRED ACTIONS – INSTRUCTIONS ARE PROVIDED FOR OPERATOR ACTIONS IN WO-198950. THE ACTION TO NOTIFY THE CONTROL ROOM IF POOL WATER LEVEL IS LOWERING IS SIMPLE AND NOT COMPLICATED.

5. SPECIFIC OPERATOR TRAINING NECESSARY TO CARRY OUT ACTIONS, INCLUDING ANY OPERATOR QUALIFICATIONS REQUIRED TO CARRY OUT ACTIONS – WO-198950 PROVIDES INSTRUCTIONS FOR THE FUEL POOL LEVEL MONITOR TO RECEIVE JUST IN TIME TRAINING. OPERATORS WHO ARE INFORMED BY THE FUEL POOL LEVEL MONITOR ARE QUALIFIED OPERATIONS PERSONNEL WHO WILL USE EXISTING PROCEDURES FOR ANNUNCIATOR RESPONSE AND OFF NORMAL EVENTS AS THEY ARE NORMALLY TRAINED TO IMPLEMENT.

6. ANY ADDITIONAL SUPPORT PERSONNEL AND/OR EQUIPMENT REQUIRED BY THE OPERATOR TO CARRY OUT ACTIONS – WORK ORDER WO-198950 ENSURES THAT A FUEL POOL LEVEL MONITOR IS ESTABLISHED TO COMPLY WITH TEMPORARY MODIFICATION EC 18232. THE MONITOR CAN CONTACT THE CONTROL ROOM BY RADIO OR PHONE.

7. A DESCRIPTION OF INFORMATION REQUIRED BY THE CONTROL ROOM STAFF TO DETERMINE WHETHER SUCH OPERATOR ACTION IS REQUIRED, INCLUDING QUALIFIED INSTRUMENTATION USED TO DIAGNOSE THE SITUATION AND TO VERIFY THAT THE REQUIRED ACTION HAS BEEN SUCCESSFULLY TAKEN – WO-298950 PROVIDES INSTRUCTIONS FOR OPERATORS TO MANIPULATE TEMPORARY SWITCHES INSTALLED BY TEMPORARY MODIFICATION EC 18232 WHICH WILL ENSURE THAT THE FUEL POOL PUMPS ARE MAINTAINED IN A RUNNING STATUS UNLESS NOTIFIED THAT FUEL POOL LEVEL IS DROPPING, AT WHICH TIME OPERATORS MAY SECURE THE FUEL POOL PUMPS VIA CONTROL SWITCHES IN THE CONTROL ROOM THAT HAVE ON/OFF INDICATION.

8. THE ABILITY TO RECOVER FROM CREDIBLE ERRORS IN PERFORMANCE OF MANUAL ACTIONS AND THE EXPECTED TIME REQUIRED TO MAKE SUCH A RECOVERY – THE MANUAL ACTIONS ASSOCIATED WITH THIS TEMPORARY MODIFICATION ARE TO MANUALLY TRIP THE FUEL POOL PUMPS SHOULD THE FUEL POOL LEVEL DROP TO LOW LOW LEVEL. THIS ACTION REQUIRES THE MANIPULATION OF CONTROL SWITCHES IN THE CONTROL ROOM TO TURN THE FUEL POOL PUMPS ON AND OFF AS APPROPRIATE. THIS REQUIRES VERY LITTLE TIME TO PERFORM OR TO CORRECT IF DONE IN ERROR. FUEL POOL TEMPERATURE RISE DOES NOT OCCUR RAPIDLY ON LOSS OF FUEL POOL COOLING SO THERE IS MORE THAN SUFFICIENT TIME TO RESTART THE PUMPS IF THEY WERE INADVERTENTLY TRIPPED.

9. CONSIDERATION OF THE RISK SIGNIFICANCE OF THE PROPOSED OPERATOR ACTIONS – THE OPERATOR ACTIONS TO SECURE THE FUEL POOL PUMPS ON LOW LOW LEVEL IN THE SPENT FUEL POOL IS TO PROTECT THE PUMPS FROM DAMAGE SO THAT THEY WILL BE AVAILABLE AS SOON AS LEVEL IS RESTORED IN THE SPENT FUEL POOL. BY DESIGN THE SPENT FUEL POOL LEVEL CANNOT DROP BELOW 40 FT 6 IN WHICH ENSURES THAT THE SPENT FUEL REMAINS COVERED AND IS PROVIDED COOLING. OPERATIONS PROCEDURE OP-901-513 PROVIDES ADEQUATE INSTRUCTIONS TO ADDRESS A MALFUNCTION OF THE SPENT FUEL POOL COOLING SYSTEM.

THE NET RESULT IS THAT THERE IS LESS THAN A MINIMAL INCREASE IN THE PROBABILITY OF FUEL POOL PUMP FAILURE DUE TO ADMINISTRATIVE CONTROLS TO REPLACE THE AUTOMATIC PUMP TRIP ON LOW LEVEL WHICH IS CONSIDERED ACCEPTABLE.

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

BASIS: THERE IS NO ACCIDENT PREVIOUSLY EVALUATED IN THE UFSAR FOR WHICH THE CONSEQUENCES WOULD BE AFFECTED AS A RESULT OF BYPASSING THE SPENT FUEL POOL PUMPS LOW LEVEL PUMP TRIP. THE PUMPS ARE NOT A FACTOR IN THE FUEL HANDLING ACCIDENT DOSE RELEASE ANALYSIS SO THERE IS NO IMPACT.

A LOSS OF FUEL POOL INVENTORY DUE TO A BREAK IN THE FUEL POOL COOLING SYSTEM PIPING IS EVALUATED IN THE UFSAR WHICH RESULTS IN THE LOSS OF BOTH FUEL POOL COOLING PUMPS ON LOW LEVEL. SIPHON BREAKERS PREVENT FUEL POOL LEVEL FROM DROPPING BELOW 40 FT 6 IN FOR THIS ACCIDENT. NO CREDIT IS ASSUMED FOR THE PUMP TRIP ON LOW LEVEL. AS A RESULT, THE FUEL POOL SYSTEM PIPE BREAK BOUNDS THE CONDITION WHERE BOTH FUEL POOL PUMPS WOULD FAIL AS A RESULT OF LOW FUEL POOL LEVEL. ADDITIONALLY, SHOULD FUEL POOL LEVEL DROP TO 40 FT 6 IN, BOTH FUEL POOL PUMPS WOULD LOSE FLOW RESULTING IN A "FUEL POOL PUMPS DISCH PRESS LO" ALARM IN THE CONTROL ROOM, AT WHICH TIME OPERATORS SHOULD MANUALLY TRIP BOTH FUEL POOL PUMPS TO PREVENT PUMP DAMAGE.

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

BASIS: THE FUEL POOL PUMPS TAKE SUCTION FROM THE SPENT FUEL POOL AT ELEVATION +40' 6". EVEN IF LEVEL BEGINS DROPPING AND THE FUEL POOL PUMPS ARE NOT MANUALLY SECURED BY OPERATIONS, THE PUMPS CANNOT PUMP THE LEVEL DOWN BELOW 40' 6". WATER CANNOT BE SIPHONED BELOW ELEVATION +40' 6" FROM THE DISCHARGE OF THE FUEL POOL PUMPS BECAUSE THERE ARE SIPHON BREAKERS IN THE FUEL POOL PUMPS DISCHARGE LINES ALSO AT THE +40' 6" ELEVATION. IF SUCTION FLOW IS LOST TO THE FUEL POOL PUMPS, THEN IT FOLLOWS THAT DISCHARGE PRESSURE WILL DROP BELOW NORMAL RESULTING IN A "FUEL POOL PUMPS DISCH PRESS LO" ALARM IN THE CONTROL ROOM. OPERATORS WOULD THEN ENTER OFF NORMAL PROCEDURE OP-901-513, SPENT FUEL POOL COOLING MALFUNCTION, AND ATTEMPT TO RESTORE FUEL POOL COOLING. ADDITIONALLY, IF THE FUEL POOL TEMPERATURE RISES TO 135 °F, A "FUEL POOL TEMPERATURE HI" ANNUNCIATOR WILL ALARM IN THE CONTROL ROOM. IN THE UNLIKELY EVENT THAT ALL COOLING IS LOST TO THE SPENT FUEL STORAGE POOL, IT WOULD TAKE 2.89 HOURS FOR THE BULK POOL TEMPERATURE TO RISE FROM 152 °F TO 212 °F ASSUMING A FULL CORE OFFLOAD STARTING THREE DAYS AFTER SHUTDOWN. THIS TIME ALLOWS SUFFICIENT TIME FOR OPERATORS TO INTERVENE AND LINE UP AN ALTERNATE SOURCE OF REPLENESHING THE POOL INVENTORY AND REMOVING DECAY HEAT. THIS OUTCOME IS NOT CHANGED AS A RESULT OF BYPASSING THE FUEL POOL PUMPS TRIP CONTACTS PER TEMPORARY MODIFICATION EC-18232.

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

BASIS: LOSS OF SPENT FUEL POOL COOLING IS ANALYZED IN THE UFSAR. A FUEL POOL COOLING SYSTEM PIPE BREAK IS ANALYZED WHICH WOULD RESULT IN A LOSS OF BOTH FUEL POOL COOLING PUMPS. THIS TEMPORARY MODIFICATION BYPASSES AN AUTOMATIC TRIP WHICH WOULD PROTECT THE FUEL POOL COOLING PUMPS FROM DAMAGE IN THE EVENT OF LOW LOW FUEL POOL LEVEL OF 40 FT 6 IN. FAILURE TO TRIP THE FUEL POOL PUMPS ON LOW LOW LEVEL IS BOUNDED BY THE ANALYZED FUEL POOL COOLING SYSTEM PIPE BREAK ACCIDENT SINCE PUMP FLOW STOPS WHEN LEVEL DROPS TO 40 FT 6 IN DUE TO THE SYPHON BREAKERS.

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

BASIS: BYPASSING THE FUEL POOL PUMPS LOW LEVEL TRIP CONTACTS RESULTS IN THE PUMPS NOT TRIPPING ON LOW LOW SPENT FUEL POOL LEVEL AND WILL NOT GENERATE AN ALARM IN THE CONTROL ROOM AS WOULD NORMALLY BE THE CASE PER EXISTING DESIGN. SINCE ADMINISTRATIVE CONTROLS WILL BE ESTABLISHED TO NOTIFY OPERATIONS OF DROPPING SPENT FUEL POOL LEVEL, THERE WILL BE AMPLE TIME FOR OPERATORS TO SECURE THE FUEL POOL PUMPS IF NECESSARY. THE FUEL POOL PUMPS MAY BE EASILY STARTED AND STOPPED FROM THE CONTROL ROOM. A LOSS OF BOTH FUEL POOL COOLING PUMPS WOULD BE NO WORSE THAN A BREAK IN THE COOLING PUMPS SUCTION LINE WHICH IS EVALUATED IN THE UFSAR. THE FUEL POOL SYSTEM FAILURE MODES AND EFFECTS ANALYSIS IN UFSAR TABLE 9.1-4 CREDITS THE DESIGN OF THE SYSTEM WHICH INCLUDES SIPHON BREAKERS AND SUCTION LINE AT ELEVATION JUST BELOW THE NORMAL WATER LINE WHICH PREVENTS SIPHONING BELOW ELEVATION +40' 6".

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

BASIS: THE FISSION PRODUCT BARRIER THAT SHOULD BE CONSIDERED HERE IS THE FUEL CLADDING BARRIER OF SPENT FUEL IN THE SPENT FUEL POOL. EVEN IF BOTH FUEL POOL COOLING PUMPS ARE NOT AVAILABLE, THE DESIGN OF THE FUEL POOL COOLING SYSTEM PRECLUDES THE POSSIBILITY OF SIPHON DRAINING THE POOL AND PROVIDES TWO SEPARATE SOURCES OF SEISMIC CATEGORY 1 MAKEUP WATER SYSTEMS. WATER LEVEL WILL REMAIN ABOVE THE TOP OF THE FUEL TO PROVIDE ADEQUATE COOLING AND ENSURE THE FUEL CLADDING BARRIER LIMITS ARE NOT EXCEEDED.

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

BASIS: A METHOD OF EVALUATION DESCRIBED IN THE UFSAR IS NOT AFFECTED BY THIS CHANGE.

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

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I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # / Rev. #:** 2011-02 / Rev 0

Proposed Change / Document: ECN 25944, Changes to EC 14765 for Calculation MPR-2390 Revision 3 SDC Gas Intrusion Analysis

Description of Change:

The main ECN25944 and Calculation ECM03-003 (MPR2390 R-3) "Analysis of Waterford Station, Unit 3 SDC System Gas Accumulation" changes were to revise the bypass piping models to update the solenoid bypass valve flow coefficient to C_v = 5.0 gpm/psi and add a 0.35 inch diameter orifice upstream of the SI-4052A(B) bypass valve in the high pressure bypass piping.

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

The ECM03-003 calculation revision 0 supported the conclusions authorized in EC 14765; the revision of ECM03-003 warrants review of conclusions previously supported in revision 0 of the calculation. The ECM03-003 revision 0 calculation information was previously evaluated under EC 14765 50.59, number 2010-006 revision 0.

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

- The gas void in the SDC suction piping will not affect LPSI system operation since the gas void will remain isolated with the system in service.
- For normal shutdowns (non design basis events), the gas void at the system high point will require venting at valve SI-4051A(B) prior to opening valve SI-407A(B) and a LPSI pump start when reactor coolant system (RCS) pressure is less than 85 psia (with instrument uncertainty - 100 psia PMC / 110 psia board indicator) to meet the NEI guidance.
- Design basis accident conditions will not allow venting therefore further analysis has been performed in ECM03-003 which concludes the LPSI pumps flow may degrade to 71% of the desired 4100 gpm flowrate for approximately two minutes as the gas passes through the pump. The flow rate will return to the desired 4100 gpm without damage or air binding of the pump.
- For a failure of bypass / equalization solenoid valves SI-4052A(B) to open, valves SI-405A(B) can be opened with RCS pressure below 230 psia (with instrument uncertainty: 213 psia PMC / 203 psia board indicator) without causing a pressure transient to lift relief valves SI-406A(B) or cause excessive dynamic loads on the piping.
- The RCS system leak rate through valves SI-401A(B), SI-405A(B) and SI-4052A(B) needs to be less than 0.26 gal/min (train A) and 0.28 gal/min (train B). The existing programs for void detection and reactor coolant leakage will ensure meeting this requirement. OP-903-026 "Emergency Core Cooling System Valve Lineup Verification", requires inspections below SI-407A(B) for void detection at 31 day intervals. If a void is discovered the required UT will ensure the void is within the evaluated size. OI-040-000 "Reactor Coolant System Leakage Monitoring" requires investigation when unidentified leakage exceeds 0.1 gpm. A statement will be added to attachment 6.6 "An unidentified leak rate value greater than 0.26 gpm in conjunction with continued voiding below valve SI-407A(B) may be indicative of leakage past SI-401A(B).

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

- OP-903-026 R-17 "Emergency Core Cooling System Valve Lineup Verification" will require revision of sections 7.6 and 7.7 to remove the introduction of a gas void requirement as the SI-4052A(B) bypass valve will prevent the hydraulic transient and relief valve lifting. The water levels reflected in sections 6.11 and 6.12 should be revised to reflect the acceptable void sizes for system operability determination. Per ECM03-003 (MPR-2390 R-3) Sect. 3.2.1 using the conservative RWSP 55 degree temperature, Train A water level (gas void) may extend 21.5 ft. below the disc of valve SI-407A, Train B water level (gas void) may extend 25.1 ft. below the disc of valve SI-407B.
- Isometric drawing ESSE-SI205 and ESSE-SI206 along with flow diagram G-167 sheet 2 are revised to reflect the addition of an orifice coupling upstream of valve SI-4052A(B) to assure the bypass piping flow rate and resultant downstream pressure transient remains below the set point of the LTOP relief valves SI-406A(B). This flow control / limit results in a 10 minute fill / equalization time limit.

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

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

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

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

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

The base Engineering Change (EC) 14765 authorizes the installation of the shutdown cooling (SDC) system fill / pressure equalization bypass line and solenoid valves SI-4052A(B). The Process Applicability Determination (PAD) and 10CFR50.59 (#2010-06 Rev 0) for EC 14765 reviewed and evaluated the installation of the SDC fill / pressurization equalization bypass line and addition of valves SI-4052A(B). The supporting calculation ECM03-003 revision 0 supported the authorization approved in EC 14675. The revision of the ECM03-003 calculation to revision 1 necessitates the evaluation of changes made to the supporting calculation. The authorization of ECN 25944 is the notification mechanism ensuring the review of relevant conclusions in EC 14765 remains intact.

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

Preparer: James W Tinkle / see ECN 25944 endorsement / Iepson Engineering / 03-31-11
Name (print) / Signature / Company / Department / Date

Reviewer: William J Steelman /see ECN25944 endorsement / EOI WF3 Licensing / 03-31-11
Name (print) / Signature / Company / Department / Date

OSRC: Steven T Adams / see ECN25944 endorsement / EOI WF3 Director NSA / 03-31-11
Chairman's Name (print) / Signature / Date

WF3 11-06
OSRC Meeting #

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II. 50.59 EVALUATION

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

Does the proposed Change:

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

   BASIS:

   The ECN25944 modification change is designed to the same safety classifications as the existing system so the frequency of a pipe break while on Shutdown Cooling (SDC) has not increased. Specifically, the 0.35" diameter orifice to the SDC bypass fill / pressurization line around the SI-405A(B) valve is designed in accordance with ASME Section III class 1 and class 2. Applying the same piping code for the orifice as the adjacent SDC piping minimizes the increase in frequency of occurrence of accidents evaluated in the Waterford 3 UFSAR.

   The SDC system bypass fill / pressurization orifice is installed in the high pressure piping side upstream of valve SI-405A(B) and is isolated downstream from the lower pressure SDC LPSI piping by SI-4052A(B) solenoid valve. During normal plant operations the upstream side of the orifice is isolated from the RCS by closure of the SI-401A(B) valve; the SI-401A(B) valve is tested for RCS leakage through the valve seat when closed. The SDC system is not generally considered an accident initiator due to the isolation of the SDC until cold shutdown conditions are attained at which time the SDC can be placed into service.

   The SDC suction gas accumulation is not caused by this modification. ECN25944 and Calculation ECM03-003 (MPR2390 R-3) only provides revised requirements to minimize the consequences of the already existing condition. Thus, the recommended actions do not increase the frequency of an accident described in the UFSAR.

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

   The frequency of accident occurrence is not increased with the addition of the orifice to the SDC bypass fill line by the application of the existing piping ASME Section III code requirements combined with the SDC line the bypass orifice is installed in being isolated from the RCS during normal power. The frequency of accidents described in the Waterford 3 UFSAR and the plant responses are unchanged by the addition of the bypass orifice.

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

   BASIS:

   The ECN25944 modification is designed to the same safety classifications as the existing system so the likelihood of a pipe break during SDC has not increased. Specifically, the 0.35" diameter orifice to the SDC bypass fill / pressurization line around the SI-405A(B) valve is designed in accordance with ASME Section III class 1 and class 2. This modification applies the same safety classifications of the SDC system to the bypass orifice so the likelihood of a pipe break when the SDC system is in service has not increased. The Low Pressure Safety Injection (LPSI) SDC system is not aligned for SDC mode above 350F and 392 psig [UFSAR 6.3.2.9.7] plant mode 4, or below, and the orifice does not challenge plant operation with Reactor Coolant System (RCS) above this temperature and pressure.

   The performance of the LPSI SDC system from a gas void between SI-405A(B) /SI-4052A(B) /SI-406A(B) and SI-407A(B) is evaluated in the ECM03-003 (MPR2390 R-3) calculation. The LPSI SDC tolerance to

gas voiding and LPSI pump A(B) performance with administratively controlled gas volumes along with bounding LOCA operation gas volumes is presented in ECM03-003 (MPR2390 R-3). The ECM03-003 (MPR2390 R-3) calculation provides system capability clarification with regards to operation and LTOP tolerance that does not change the existing UFSAR likelihood of a structure, system, or component important to safety malfunction, and does not change the conclusions reached utilizing ECM03-003 revision 0. The existing two train design of SDC permits placement of the alternate SDC train into service should the SI-405A(B) valve fail to operate. The ECM03-003 (MPR2390 R-3) calculation concludes SDC can be placed into service by reducing RCS pressure to less than 230 psia and opening the SI-405A(B) valve should the SI-4052A(B) valve fail to open.

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

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

BASIS:

The SDC suction gas accumulation is not caused by this modification. ECN25944 and Calculation ECM03-003 (MPR2390 R-3) only enhance the existing modification and provide revised requirements to minimize the consequences of the already existing condition. The impact on accident consequences have not been increased by more than a minimal amount. The orifice addition and the ECM03-003 (MPR2390 R-3) results with respect to the pressure transient, gas transport, RCS leakage requirement, and the gas void dynamic venting will be addressed below.

The orifice installation in the bypass fill ¾" line around valve SI-405A(B) is not aligned for LPSI SDC mode above 350F and 392 psig [UFSAR 6.3.2.9.7] plant mode 4. The loss of ¾" pipe integrity at the orifice with the LPSI in SDC service where the redundant train of SDC could be placed into service are bounded by existing analyses in the UFSAR for both large break and small break LOCAs. The orifice is designed to the same ASME Section III class 1 and 2 material requirements as the bypass SDC system piping that does not change the consequences of the existing accidents previously evaluated in the UFSAR. The bypass orifice is upstream of the containment isolation valve SI-4052A(B) and as such does not impact the containment isolation of the SDC penetrations 40 and 41.

The existing administrative controls limit the size of potential gas voiding in the SDC reactor containment penetrations 40 and 41. The ECM03-003 (MPR2390 R-3) does not change conclusions reached from the revision 0 of the calculation.

The ECM03-003 (MPR2390 R-3) gas transport analysis includes a bounding gas void passing through the SDC LPSI pumps for the limiting case that a gas void develops in the SDC reactor containment penetration piping coincident with a LOCA, or small break LOCA requiring placement of the SDC into service without venting the gas void. The plant accidents were reviewed and found to be bounded by the LOCA conditions; these included plant accidents such are the main steam line break and steam generator tube rupture. The minor reduction of LPSI flow during the initial two minutes of placing the SDC system into service with the maximized gas void with an RCS pressure as low as 70 psia is analyzed to demonstrate the gas void does not gas bind the LPSI pump, though the maximized gas void would exceed the 5% gas void fraction NEI gas void guidance that is the current industry accepted limit. Administrative controls are in place to preclude the bounding gas void from forming. Should a bounding gas void begin to form this would represent a degraded condition; this condition would be tracked in the 10CFR50 Appendix B Corrective Action Program and is outside of the 10CFR50.59 process.

The ECM03-003 (MPR2390 R-3) pressure transient analysis requires the SDC system placement into service with an additional ten minute gas void pressurization prior to opening the SI-405A(B) valve. This

delay in SDC service will not cause more than a minimal increase in the consequences of accidents evaluated in the UFSAR. With a SI-4052A(B) single failure, the other redundant train will remain available and capable of meeting the required safety function. ECM03-003 (MPR2390 R-3) evaluated that the failed SI-4052A(B) train may be placed in service below 230 psia with no pressure transient threat.

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

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

Thus, this modification does not result in more than a minimal increase the consequences of an accident previously evaluated in the UFSAR.

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

BASIS:

The installation of the bypass orifice to the SDC system will not affect the system's previously evaluated malfunction consequence as part of the reactor coolant pressure boundary as the orifice is designed to the same safety classifications as the SDC system. The orifice is isolated from the RCS pressure when the RCS is above 350F and 392 psig. The RCS pressure isolation is provided by valve SI-401A(B) being closed during normal plant operation with the RCS above 350F and 392 psig. The capability to equalize pressure across valve SI-405A(B) prior to opening SI-405A(B) while placing SDC system into service does not affect the results of the existing, and bounding UFSAR LOCA of the existing 14" shutdown cooling piping. The existing redundant trains of SDC provide alternate means of SDC should the bypass orifice become blocked, or if the SI-4052A(B) valve failed to open. The likelihood of the bypass orifice becoming blocked is extremely remote due to the orifice 0.35" diameter and the RCS is maintained at a high degree of cleanliness though the monitoring of chemistry for detection of degradation material in the RCS combined with the charging system feed-and-bleed through filtered pressurizer spray and letdown through the chemical volume control system.

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

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

BASIS:

The types of accident that could occur would be from the loss of ¾" pipe integrity at the orifice with the LPSI in SDC service. The LPSI SDC system is isolated when the RCS is at service above 350F and 392 psig by valve SI-401A(B). A loss of pipe integrity at the orifice with the RCS at or above this temperature and pressure does not pose an accident other than previously evaluated; there is no flow in the isolated line other than presumed leakage through the SI-401A(B) valve would leak through-wall and collect in the containment sump. Leakage through the SI-401A(B) valve is limited and measured as part of RCS leakage, and would not present a challenge to the sump collection capacity. At temperature and pressures below the 350F and 392 psig, when LPSI SDC is aligned, the loss of ¾" pipe integrity is bounded by the existing small line LOCA accident evaluation.

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

The blockage failure of both bypass orifices in the SDC "A" and "B" trains [similar to the failure of both of the SI-4052A and SI-4052B valves to open] is shown analytically to not prevent putting SDC into service through use of bounding maximized gas voids with an RCS pressure as low as 70 psia while demonstrating the maximized gas void does not gas bind the LPSI pumps. The simultaneous failure of both bypass loops match the ECM03-003 revision 1 LOCA service with conditions of a maximized gas void that includes the P40 / P41 reactor containment penetration SDC piping between SI-405A(B) and SI-407A(B) combined with gas voids downstream of the SI-407A(B) valves. The analysis concludes the LPSI pumps wound not be gas bound, and would transport the maximized gas void within two minutes of being placed into operation. The reduced LPSI pump flow for this initial two minutes is within the operator time to energize the motive driver for SDC valves SI-405A(B) and SI-407A(B), and as such does not create the possibility of an accident of a different type than any previously evaluated in the existing UFSAR. The Waterford 3 plant administratively controls the size of the potential SDC gas void through OP-903-026 instructions that vent, or measure gas voids in the SDC piping downstream of SI-407A(B). To limit the accumulation of gas voids in the reactor containment penetration P40 and P41 SDC piping the leakage value of 0.26 gallons per minute was determined by analysis as acceptable and is bounded by the existing OP-040-000 investigative action at 0.1 gpm for RCS unidentified leakage. Through the administrative control of monitoring RCS leakage through valve SI-401A(B) the potential of gas accumulation in the reactor containment penetration P40 and P41 SDC piping is minimized to a LPSI pump tolerable condition for pump function. In addition the administrative control of OP-903-026 abnormal gas accumulation downstream of SI-407A(B) is identified and controlled through the condition report process. No common failure mechanism is being caused, rather controls and limits will identify and preclude a possible "unknown" plant condition that would place both redundant SDC trains beyond analyzed condition [with regards to gas voiding].

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

(1) The specific operator actions required;

Specific operator action to close the breakers for valves SI-405A(B) and SI-4052A(B) is similar to those actions identified in OP-009-005 for closing breakers for the SI-401A(B) valves. No adverse impact.

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

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

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

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

(4) The procedural guidance for required actions;

Existing procedural guidance for closure of SI-401A(B) valve breaker provides adequate template for operator guidance in the closure of the breaker for SI-405A(B) and SI-4052A(B) valves. No adverse impact.

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

Operator training to perform specific action to close the breaker for SI-405A(B) and SI-4052A(B) valves is the same as the to the risk presently evaluated for SI-401A(B) breaker closure. No adverse impact.

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

The same operations personnel and equipment that address the closure of the SI-401A(B) valves can close the breaker switches for valves SI-405A(B) and SI-4052A(B). No adverse impact.

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

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

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

The ability to recover from inadvertent action to close the breaker for SI-405A(B) and SI-4052A(B) valves is similar to the recovery presently evaluated for SI-401A(B) valve breaker closure. The recovery time is enveloped in the recovery time for closure of the SI-401A(B) valve breakers. No adverse impact.

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

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

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

The Shutdown Cooling system is not required for being placed into service at the beginning of a plant event. The later time sequence in the event for SDC system to be placed into service affords the time for operator action to manually close the breaker for SI-405A(B) and SI-4052A(B) valves. The time response is similar to the time evaluated for SI-401A(B) operator breaker closure action. No adverse impact.

No new accident initiator or types are created by this change as the bypass orifice meets the SDC system necessary code and design requirements.

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

BASIS:
The previously experienced SDC system pressure transient and LTOP relief valve lift was a result of a rapid compression of an accumulated gas void in the SDC piping. The bypass line orifice addition will control the void compression and limit the transient pressure below the relief valve setpoint ensuring the system design and licensing basis is maintained. A malfunction of the bypass orifice could result in a loss of coolant and would lead to the same results that have been previously analyzed in the UFSAR for the 14" SDC piping system when the SDC is open to the RCS; SI-401A(B) valve is closed during normal plant operation and is only open when the RCS is below 350F and 392 psia. The new configuration will also not affect shutdown cooling conditions. No malfunction of the added piping or valves could lead to a different result than has been previously evaluated in the UFSAR.

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

The blockage failure of both bypass orifices in the SDC "A" and "B" trains [similar to the failure of both of the SI-4052A and SI-4052B valves to open] is shown analytically to not prevent putting SDC into service through use of bounding maximized gas voids with an RCS pressure as low as 70 psia while demonstrating the maximized gas void does not gas bind the LPSI pumps. The simultaneous failure of both bypass loops match the ECM03-003 revision 1 LOCA service with conditions of a maximized gas void that includes the P40 / P41 reactor containment penetration SDC piping between SI-405A(B) and SI-407A(B) combined with gas voids downstream of the SI-407A(B) valves. The analysis concludes the LPSI pumps would not be gas bound, and would transport the maximized gas void within two minutes of being placed into operation. The reduced LPSI pump flow for this initial two minutes is within the operator time to energize the motive driver for SDC valves SI-405A(B) and SI-407A(B), and as such does not create the possibility of a malfunction of a SSC different type than any previously evaluated in the existing UFSAR. The Waterford 3 plant administratively controls the size of the potential SDC gas void through OP-903-026 instructions that vent, or measure gas voids in the SDC piping downstream of SI-407A(B). To limit the accumulation of gas voids in the reactor containment penetration P40 and P41 SDC piping the leakage value of 0.26 gallons per minute was determined by analysis as acceptable and is bounded by the existing OP-040-000 investigative action at 0.1 gpm for RCS unidentified leakage Through the administrative control of monitoring RCS leakage through valve SI-401A(B) the potential of gas accumulation in the reactor containment penetration P40 and P41 SDC piping is minimized to a LPSI pump tolerable condition for pump function. In addition the administrative control of OP-903-026 abnormal gas accumulation downstream of SI-407A(B) is identified and controlled through the condition report process. The operation of the SDC LPSI remains within the design parameters of the plant SSC and does not create a malfunction with different results than any previously evaluated in the UFSAR.

No possibility for a malfunction of a SSC important to safety is created by this change as the bypass orifice meets the SDC system necessary code and design requirements.

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

BASIS: The orifice installation in the bypass fill ¾" line around valve SI-405A(B) does change the reactor coolant pressure boundary when the RCS is below 350F and 392 psig. The orifice has been designed to meet the requirements of ASME Section III for class 1 and 2 piping to meet the fission product barrier requirements applied to the SDC system. The bypass orifice is located upstream of containment isolation valve SI-4052A(B). Containment integrity and fuel cladding fission barriers remain unchanged from the bypass orifice installation. The orifice is isolated from the RCS inventory by valve SI-401A(B) and any

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leakage through a loss of pipe integrity would be as a result of leakage through the SI-401A(B) valve. The SI-401A(B) valve is shut with a leakage verification through the valve in combination with logic that prevents opening the SI-401A(B) valve with RCS at normal operating pressure. The accidents analyzed for loss of fission product barrier for the small break LOCA in the existing UFSAR bound the potential failure of the orifice ¾" pipe integrity due to the SDC system being isolated with valve SI-401A(B) from the RCS during normal plant operation. The loss of bypass orifice integrity results in a loss of coolant when the SDC system is in operation with the RCS is below 350F and 392 psig. The bypass orifice installation will also not affect shutdown cooling conditions for the plant. The use of the same safety classifications as the existing SDC system would lead to the same results that have been previously analyzed in the UFSAR.

The ECM03-003 calculation revision 1 clarifications do not change or result in a change to the fission product barrier. The conclusions determined from ECM03-003 revision 0 calculation remain unchanged with the exception of the inclusion of the orifice as a fission product barrier in revision 1 of the calculation.

No malfunction of the added bypass orifice could lead to different result than has been previously evaluated in the UFSAR.

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

BASIS: The analysis performed in support of the installation of the orifice installation in the bypass fill ¾" line around valve SI-405A(B) are consistent with the existing UFSAR described methodologies. The proposed change is limited in scope to the LPSI SDC system and the previously approved methods were used to evaluate the change to the reactor coolant pressure boundary without departure from existing design bases and safety analysis.

The ECM03-003 calculation revision 1 analysis methodology is not explicitly described in the UFSAR and did not change or result in a change to those methods of evaluation described in the existing UFSAR.

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

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I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # / Rev. #:** 2011-03

Proposed Change / Document: EC26496

Description of Change:
This 50.59 Evaluation will address the potentially adverse impacts of the new Emergency Operating Procedure contingency actions to respond to the single failure of a Low Pressure Safety Injection (LPSI) pump to automatically stop when a Recirculation Actuation Signal (RAS) is received.

The Low Pressure Safety Injection System is designed to inject borated water into the Reactor Coolant System (RCS) to flood and cool the reactor core and to provide for heat removal from the reactor core for extended periods following a Loss of Coolant Accident (LOCA). The LPSI pumps start automatically upon receipt of a SIAS. Pump suction is supplied by the RWSP. Once the RWSP reaches 10% indicated level, a RAS is generated which automatically stops all operating LPSI pumps.

As part of the Waterford 3 response to Generic Letter 2004-02, the NRC required licensees to address a single failure in which a Low Pressure Safety Injection (LPSI) pump fails to automatically stop when a RAS is received. A RAS is generated when the Refueling Water Storage Pool (RWSP) level reaches 10% indicated level and switches ECCS/CS suction from the RWSP to the Safety Injection Sump. This failure is limiting for sump design as it results in increased flow through the Safety Injection Sump that is common to both ECCS/CS trains. The typical limiting failure assumed in most accident analysis, loss of a DC buss with subsequent loss of a protective train, is not limiting as it would result in a decrease in total sump flow. The design flow rate for the Safety Injection Sump is 6470 gpm based on the operation of two High Pressure Safety Injection (HPSI) and two Containment Spray (CS) pumps in run-out conditions. Failure of a LPSI pump to stop on RAS could increase the flow through the sump by approximately 5650 gpm to a total of 12120 gpm. The added flow would cause an increase in the amount of debris transported to the strainers and would also cause an increase to the head loss across the strainers. To address this potential failure, contingency actions are being added to Emergency Operating Procedure (EOP) OP-902-002 and OP-902-008. The contingency actions would be taken only if a LPSI pump does not automatically stop on RAS as designed. The first action would be to manually secure the pump via the control switch on the control panel in the main control room. The second action would only be taken if the pump continues to operate. The closed Shutdown Cooling Warm-up Valve (SI-135) associated with the operating pump would be throttled open and the Flow Control Valves (SI-138/139) which open on a Safety Injection Actuation Signal (SIAS) would then be fully closed. This would stop the increased flow through the sump while providing a temporary flow path for the pump.

As previously stated, the contingency actions being added to the EOPs would only be taken if one of the operating LPSI pumps fails to automatically stop when a RAS signal is received. Manual action would be required as the LPSI pump would begin to take suction from the Safety Injection Sump and could cause the head loss to increase such that the operating pumps in both trains could lose adequate NPSH. This is a result of the Waterford 3 common sump design. Sump strainer head loss testing, documented in report WF3-ME-10-00006, confirmed that the added flow from an operating LPSI pump would not cause vortexing or air injection with a clean strainer. During a LOCA, the strainer will be initially clean when recirculation begins. As time passes, debris will be transported in the flow streams to the strainers causing the head loss to gradually increase. Securing the operating LPSI pump within 7 minutes post RAS will prevent an increase in head loss above the design value.

The Waterford debris generation and transport calculations are not time dependent and only determine the maximum bounding generated and transported debris loads. As documented in letter W3F1-2010-0032, "Response To Request For Additional Information Regarding Final Supplemental Response To Generic Letter 2004-02, Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors (TAC No. MC4729)", if the added flow is stopped by initiating closure of the flow

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

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control valves within 7 minutes post RAS then no additional debris would transport due to the short elevated flow period.

The planned approach to resolve this issue has been discussed with the NRC GSI-191/GL04-02 Staff. The NRC Staff identified no concerns with the Waterford planned resolution.

The RAS Initiation Criteria in the EOPs currently instruct operators to "Verify that both LPSI pumps are stopped" after a RAS has been initiated. Failure of one pump to stop would be immediately identified and actions to throttle open SI-135 and close SI-138/139 could be taken quickly. Once SI-135 is opened, the added flow through the sump would immediately begin to decrease as the pump flow takes the less resistant path through the warm-up line. Closing SI-138/139 would ensure no added flow passes through the sump by isolating the operating LPSI train from containment.

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

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

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

Preparer: Gregory N. Ferguson / See EC 26496 / EOI / DE-Civil / 4-28-11
Name (print) / Signature / Company / Department / Date

Reviewer: Dale Gallodoro / See EC 26496 / EOI / Design Engineering / 4-28-11
Name (print) / Signature / Company / Department / Date

OSRC: Steven Adams / See EC 26496 / EOI / NSA / 4-28-11
Chairman's Name (print) / Signature / Date

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II. 50.59 EVALUATION

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

Does the proposed Change:

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

   BASIS:

   The proposed change does not affect any accident initiator. UFSAR Chapter 15 was reviewed to identify which accidents previously evaluated could be initiated or caused by the proposed change. UFSAR section 15.5.1.2 discusses the inadvertent operation of the Emergency Core Cooling System (ECCS) with respect to an increase in Reactor Coolant System Inventory. The conclusion of this section is that due to the pressure limitations of the ECCS, an inadvertent increase in Reactor Coolant System Inventory due to ECCS malfunction is not possible during power operation. The proposed change will not affect this conclusion. The possibility of initiation of an intersystem LOCA or any other accident potentially initiated by the Safety Injection System is not affected by the proposed change as it only affects system response. The proposed EOP contingency actions would only be taken during an accident and in response to a failed existing automatic action. The proposed change also does not create any new system interactions and has no impact on operation or function of any system or equipment that in any way could cause an accident. Therefore, the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

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

   BASIS:

   The postulated failure of a LPSI pump to trip on a RAS is being added to the UFSAR Table 6.3-1, Failure Modes and Effects Analysis – Safety Injection System, as a result of NRC letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors". The proposed new EOP contingency actions would only be taken in response to the single failure of a LPSI pump to automatically trip on RAS as currently designed. The postulated failure of a LPSI pump not to trip on a RAS affects the opposite ECCS train due to the common sump strainer. The contingency actions ensure that the single failure does not adversely impact the opposite ECCS train. Since only one LPSI pump fails to trip on RAS, the opposite train's diesel fuel 7 day capacity will not be affected. An additional failure is not required to be postulated while performing the proposed contingency actions. The proposed contingency actions will be added to the applicable operating procedures and training programs. During a test of the proposed actions in the Waterford 3 simulator, it took less than 5 minutes for completion of the actions. Use of performance data to justify the action time is allowed per ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-Related Operator Action". Therefore, the proposed change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

BASIS:

The Safety Injection System is an accident mitigating system. The proposed EOP contingency actions will protect the opposite train if a LPSI pump fails to automatically stop on RAS. By protecting the opposite operating train, the current limiting safety analysis for dose consequences, which assumes loss of a single train of safety equipment, remains bounding. Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

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

BASIS:

The proposed EOP contingency actions will protect the opposite train if a LPSI pump fails to automatically stop on RAS. This ensures that the current limiting safety analysis for dose consequences, which assumes loss of a single train of safety equipment, remains bounding. Therefore, the proposed change does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

BASIS:

The Safety Injection System is an accident mitigating system and is not an accident initiator. The proposed EOP contingency actions will protect the opposite train if a LPSI pump fails to automatically stop on RAS. The actions would only be taken during the response to an action that initiates Safety Injection. Therefore the proposed change does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR.

The manual operator action to close valves SI-138A(B) and SI-139A(B) was evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 on system operation post RAS. The ten primary attribute evaluations were weighted with the existing manual operator action located in OP-902-002 and OP-902-008 to close SI-120A(B) and SI-121A(B) after RAS (2 minutes). An additional action is added after that to close the affected LPSI flow control valves SI-138A/139A or SI-139A/139B in a specified time of 5 minutes. These times have been transmitted to the NRC in letter W3F1-2010-0032.

(1) The specific operator actions required;
Specific operator action to close the SI-138A(B) and SI-139A(B) valves is similar to other actions identified in OP-902-002 and OP-902-008 for closing SI-120A(B) and SI-121A(B) valves except that more time is available to close the SI-138A(B) and SI-139A(B). No adverse impact.

(2) The potentially harsh or inhospitable environmental conditions expected;
The SI-138A(B) and SI-139A(B) flow control valve control switches are located in the control room which is the same environment for the SI-120A(B) and SI-121A(B) valves. No adverse impact.

(3) A general discussion of the ingress/egress paths taken by the operators to accomplish functions;
The hand switches for valves SI-138A(B), SI-139A(B), SI-120A(B) and SI-121A(B) are all located on the same control panel CP-8 in the control room. Ingress and egress paths are the same to the panel for all of these valves. No adverse impact.

(4) The procedural guidance for required actions;
Procedural guidance for closure of valves SI-138A(B) and SI-139A(B) valves currently exists. These valves are manipulated manually in the control room during shutdown cooling operations and can be closed to support containment isolation. The action to close these valves in the timeframes specified above has been tested in the simulator and proven successful. These valves are also stroked every

quarter for IST testing. No adverse impact.

(5) The specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions;
The closure of valves SI-138A(B) and SI-139A(B) in the sequence stated above has already been carried out in the simulator by Operations. Operator training was performed in 2011 Cycle 2 License Operator Requalification Training (Ref. TEAR W3 2011 39). No additional training is required. This does not present additional risk over what is being currently performed. No adverse impact.

(6) Any additional support personnel and/or equipment required by the operator to carry out actions;
The operations personnel and equipment that address the current operation of SI-138A(B) and SI-139A(B) will not change. No additional support personnel or equipment are required to carry out the actions to place the operating LPSI pump on recirculation. The same operations personnel and equipment also address the current closure of valves SI-120A(B) and SI-121A(B). No adverse impact.

(7) A description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken;
The determination of whether operator action is required begins with the failure of a LPSI pump to stop after RAS. An action in the EOPs currently exists to verify LPSI pumps have stopped - no change is made and the response is immediate. Control panel pump indication or existing flow instrumentation will validate LPSI pump operation and the need for further action. Also, indication of the pump continuing to operate will be if the operator attempts to stop the pump by the control switch and it continues to operate. Success is met by initiation of flow control valve closure in the operating LPSI train within the allotted 7 minutes post RAS and the diversion of flow through the recirculation path. No adverse impact.

(8) The ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery;
The ability to recover from inadvertent action in attempting to close valves SI-138A(B) and SI-139A(B) is predicated on the remaining time to correct it. A possible error would be to close off the wrong train. It is not likely to close the flow control valves on the wrong train since they are on the opposite side of the board. However, if they are closed, that train of LPSI is expected to have operated correctly and tripped so there is no plant or system affect. This has no effect on the system safety function as no flow is expected in both trains. No adverse impact.

(9) Consideration of the risk significance of the proposed operator actions;
The risk significance for action to close valves SI-138A(B) and SI-139A(B) is similar to the risk presently evaluated for closing valves SI-120A(B) and SI-121A(B). No adverse impact.

(10) Time response as outlined in ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-Related Operator Action";
The proposed new EOP contingency actions would only be taken in response to the single failure of a LPSI pump to automatically trip on RAS and ensures that the single failure does not adversely impact the opposite operating train. The contingency actions were validated on the simulator during development, the five minute time limit is included in the procedures and operators have been trained to the procedure during LOR. Use of performance data to justify the action time is allowed per ANSI/ANS-58.8-1994. No adverse impact.

No new accident initiator or types are created by this change where SI system recirculation operation meets the necessary code and design requirements.

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

BASIS:

The failure modes and effects tables for the Safety Injection and Containment Spray systems in the UFSAR were reviewed and determined to remain valid. A new failure mode and effect is being added for the failure of a LPSI pump to automatically stop when a RAS signal is received. The proposed EOP contingency actions will prevent the trip failure in one train from adversely affecting the other train. This ensures that the current limiting safety analysis for dose consequences remains bounding. Therefore the proposed change does not create the possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR.

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

BASIS:

The proposed EOP contingency actions ensure LPSI trip failure in one train does not affect performance of the other operating train. This ensures that the current limiting safety analysis for dose consequences and fission product barrier limits remain bounding. Therefore, the proposed change does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

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

BASIS:

Currently the UFSAR does not explicitly discuss any LPSI failure to trip on RAS scenarios. The existing FSAR evaluations pertaining to the operations of the Safety Injection System assume that any operating LPSI pumps stop when a RAS is received. The postulated failure of a LPSI pump to trip on a RAS is being added to the UFSAR Table 6.3-1, Failure Modes and Effects Analysis – Safety Injection System, as a result of NRC letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors". The Waterford debris generation and transport calculations are not time dependent and only determine the maximum bounding generated and transported debris loads. As documented in letter W3F1-2010-0032, "Response To Request For Additional Information Regarding Final Supplemental Response To Generic Letter 2004-02, Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors (TAC No. MC4729)", if the added flow is stopped by initiating closure of the flow control valves within 7 minutes post RAS then no additional debris would transport due to the short elevated flow period. The planned approach to resolve this issue has been discussed with the NRC GSI-191/GL04-02 Staff. The NRC Staff identified no concerns with the Waterford planned resolution. The contingency actions would ensure that the assumption remains valid if a pump does not automatically stop as currently designed. Therefore, the proposed change does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

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

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I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 **Evaluation # 2011-04 / Rev. #:** / 0

Proposed Change / Document: Technical Specification 3.4.1.5 Operability Evaluation

Description of Change:
This 50.59 evaluation addresses the potentially adverse impact of the operator action required during the OP-903-115 (Train A Integrated Emergency Diesel Generator/Engineering Safety Features Test) surveillance with relation to placing the shutdown cooling system in service while in Mode 5 for a postulated shutdown cooling malfunction event. The specific action is to open SI-124A (LPSI PUMP A DISCHARGE ISOLATION) and is controlled by OP-903-115 Attachment 10.5 Restoration Alignment.

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

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

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

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

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

from credible errors in performance of manual actions, and the expected time required to make such a recovery; and (9) consideration of the risk significance of the proposed operator actions.

NRC REGULATORY ISSUE SUMMARY 2005-20 [Reference 3] issued a new version of NRC Inspection Manual Technical Guidance Part 9900 ITSB [Reference 4]. The NRC Inspection Manual [Reference 4] lists the following requirements for crediting an operator action.
For situations where substitution of manual action for automatic action is proposed for an operability determination, the evaluation of manual action must focus on the physical differences between automatic and manual action and the ability of the manual action to accomplish the specified safety function or functions. The physical differences to be considered include the ability to recognize input signals for action, ready access to or recognition of setpoints, design nuances that may complicate subsequent manual operation (such as auto-reset, repositioning on temperature or pressure), timing required for automatic action, minimum staffing requirements, and emergency operating procedures written for the automatic mode of operation. The licensee should have written procedures in place and personnel should be trained on the procedures before any manual action is substituted for the loss of an automatic action.

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

The manual operator action was evaluated against NRC Information Notice 97-78, NRC Regulatory Issue Summary 2005-20, and ANSI/ANS-58.8-1994 and is addressed in Question #5.

References

1. Technical Specification Amendment 230.
2. NRC INFORMATION NOTICE 97-78, CREDITING OF OPERATOR ACTIONS IN PLACE OF AUTOMATIC ACTIONS AND MODIFICATIONS OF OPERATOR ACTIONS, INCLUDING RESPONSE TIMES, October 23, 1997.
3. NRC REGULATORY ISSUE SUMMARY 2005-20 REV. 1, REVISION TO NRC INSPECTION MANUAL PART 9900 TECHNICAL GUIDANCE, OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS FOR RESOLUTION OF DEGRADED OR NONCONFORMING CONDITIONS ADVERSE TO QUALITY OR SAFETY, April 16, 2008.
4. NRC Inspection Manual Technical Guidance Part 9900 ITSB, OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS FOR RESOLUTION OF DEGRADED OR NONCONFORMING CONDITIONS ADVERSE TO QUALITY OR SAFETY
5. ANSI/ANS-58.8-1994, American National Standard Time Response Design Criteria for Safety Related Operator Actions, August 23, 1994.
6. NRC GENERIC LETTER 98-02, LOSS OF REACTOR COOLANT INVENTORY AND ASSOCIATED POTENTIAL FOR LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION, May 28, 1998.
7. W3F1-98-0161, NRC Generic Letter (GL) 98-02, Loss of Reactor Coolant Inventory And Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition, November 19, 1998.
8. CN-OA-08-5, WSES-3 LOSS OF SHUTDOWN COOLING FROM MID LOOP, December 11, 2008.
9. ECS10-002 R0, WATERFORD 3 RCS TIME-TO-BOIL DUE TO LOSDC AT VARIOUS INITIAL LEVELS AND TEMPERATURES, EC23453.

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

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

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

Preparer: William J Steelman / / EOI / Licensing / 4/29/2011
Name (print) / Signature / Company / Department / Date

Reviewer: T N Schreckengast / / EOI / Operations / 4/29/2011
Name (print) / Signature / Company / Department / Date

OSRC: STEVEN T. ADAMS / / 4/29/2011
Chairman's Name (print) / Signature / Date

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II. **50.59 EVALUATION**

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

Does the proposed Change:

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

   BASIS:

   The proposed change does not affect any accident initiator. UFSAR Chapter 6 and 15 were reviewed to identify which accidents previously evaluated could be initiated or caused by the proposed change. The lower mode events (Fuel Handling Accident, Main Steam Line Break, Subcritical CEA Withdrawal) would not be impacted by the manual action to place shutdown cooling in service. The proposed manual actions would only be taken during a shutdown cooling malfunction condition. UFSAR Section 9.3.6.3.4 (Loss of Shutdown Cooling with RCS Partially Filled) describes the requirements for the loss of shutdown cooling. The placement of shutdown cooling in service was already a manual operator action as described in UFSAR section 9.3.6.3.2, SHUTDOWN COOLING SYSTEM (RESIDUAL HEAT REMOVAL SYSTEM) Manual Actions, so this has no impact on the frequency of occurrence.

   The proposed change also does not create any new system interactions and has no impact on operation or function of any system or equipment that in any way could cause an accident. Therefore, the proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

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

   BASIS:

   The proposed manual action would only be taken in response to a shutdown cooling malfunction type event. UFSAR Section 9.3.6.3.4 describes the requirements for the loss of shutdown cooling. The proposed change does not affect the likelihood of an equipment malfunction because the placement of shutdown cooling in service was already a manual operator action as described in UFSAR section 9.3.6.3.2.

   The proposed change also does not create any new system interactions and has no impact on operation or function of any system or equipment that in any way could cause an accident. Therefore, the proposed change does not result in an increase in likelihood of occurrence of malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

BASIS:

The shutdown cooling malfunction type events are not included in UFSAR Chapter 6 or 15. The shutdown cooling malfunction event consequences would be bounded by the Fuel Handling Accident (UFSAR 15.7.3). The Waterford 3 Fuel Handling Accident (FHA) Analysis of Record (AOR) documented that the failure of 60 fuel rods is the largest number of fuel rods that could fail from the worst postulated assembly drop. UFSAR Section 9.3.6.3.4 describes the requirements for the loss of shutdown cooling. The regulatory requirement is in commitment P22641 for core uncovery time requirement to be greater than one hour. Calculation CN-OA-08-5 is the design basis calculation to validate that the core uncovery requirements are met. CN-OA-08-5 page 11 shows (Scenario #5) that greater than 2 hours to core uncovery exists. CN-OA-08-5 scenario #5 is for mid-loop conditions; the plant conditions when this action will be required are at greater inventory, lower temperature, and lower decay heat (due to new fuel) than the bounding analysis. So significantly more time is available prior to core uncovery.

The other requirement is that the calculated time for the Reactor Coolant System to boil is greater than the specified closure time (SOER 09-01 Recommendation #11). Calculation ECS10-002 provides the time to boil information. The plant configuration when the OP-903-115 is occurring, the time to boil does not exceed the time required to close the containment impairments.

The proposed operator action will ensure the opposite train of shutdown cooling will be available in a timely manner and capable of performing its specified function. Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

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

BASIS:

The proposed manual action would only be taken in response to a shutdown cooling malfunction type event. The proposed change does not increase the consequences of a malfunction because the placement of shutdown cooling in service was already a manual operator action. The operator actions to place the equipment in service has not change. The timing needed to place shutdown cooling in service will also remain within the limit established in UFSAR Section 9.3.6.3.2 due to dedicated operators stationed near the equipment.

The proposed change also does not create any new system interactions and has no impact on operation or function of any system or equipment that in any way could cause an accident. Therefore, the proposed change does not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

BASIS:

The Shutdown Cooling System is an accident mitigating system and is not an UFSAR Chapter 6 or 15 accident initiator. UFSAR Section 9.3.6.3.4 describes the loss of shutdown cooling with the RCS partially filled. The proposed manual action would only be taken in response to a shutdown cooling malfunction event. The proposed change does not alter the method of placing shutdown cooling in service because this was already a manual operator action. The operator actions to place the equipment in service has not changed. Therefore the proposed change does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR.

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

(1) The specific operator actions required;
The operator actions required to place a standby train of shutdown cooling in service is contained in procedure OP-009-005. OP-903-115 Attachment 10.5 alignment 1 closes SI-124A (LPSI PUMP A DISCHARGE ISOLATION). OP-903-115 Attachment 10.5 restoration alignment restores SI-124A. The actions required by the operator are procedurally controlled and the operators will be dedicated to ensuring SI-124A will be restored upon notification of shutdown cooling malfunction event. No adverse impact.

(2) The potentially harsh or inhospitable environmental conditions expected;
The event of interest is a shutdown cooling malfunction event in Mode 5 or 6. For this event, there are no event related elevated activities, temperatures, or pressures in the Reactor Auxiliary Building (RAB), Radiologically Controlled Area (RCA), or control room. No adverse impact.

(3) A general discussion of the ingress/egress paths taken by the operators to accomplish functions;
The majority of the OP-009-005 actions occur from the control room. The additional action is controlled by OP-903-115. The actions taken outside the control room will be in the RAB and RCA. Emergency lighting is available in all locations needed and the environment conditions (2) will not be adverse. No adverse impact.

(4) The procedural guidance for required actions;
Procedural guidance for placing shutdown cooling is service is contained in OP-009-005. The additional action is controlled by OP-903-115. No adverse impact.

(5) The specific operator training necessary to carry out actions, including any operator qualifications required to carry out actions;
Placing shutdown cooling in service is already a manual operator action that is trained on as part of operator requalification training. The operators dedicated to performing specific actions for a potential shutdown cooling malfunction will be briefed as to communications protocols, standby areas, travel routes, time limits, the required procedure steps, and actions required. No adverse impact.

(6) Any additional support personnel and/or equipment required by the operator to carry out actions;
The operations personnel and equipment that address the current operation will not change. No additional support personnel or equipment are required to carry out the actions to place the standby shutdown cooling train in service. No adverse impact.

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

The determination of whether operator action is required begins with the determination of an adverse impact on the shutdown cooling system. Off normal procedure OP-901-131 will be entered for any abnormal shutdown cooling conditions. A dedicated shutdown cooling watch will be in the control room during this evolution so the identification will be prompt and associated actions will already be pre-arranged. No adverse impact.

(8) The ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery;
The ability to recover from creditable error will be immediately identified by the control room operators based upon system flow and temperature indications. The placement of shutdown cooling system is already a manual operator action so this change will not affect the credible errors that already existed. No adverse impact.

(9) Consideration of the risk significance of the proposed operator actions;
The risk significance for placing shutdown cooling in service has not changed. The process of placing the standby shutdown cooling train in service was already a manual operator action. No adverse impact.

(10) *Time response as outlined in* ANSI/ANS-58.8-1994, "*Time Response Design Criteria for Safety-Related Operator Action*";
ANSI/ANS-58.8-1994 provide time requirements for different accident scenarios. For a shutdown cooling malfunction using the ANSI/ANS-58.8-1994 guidance 10 minutes would be required for identification and diagnosis with an additional 1 minute per each required operator action. 30 minutes is also listed in ANSI/ANS-58.8-1 for operator actions required outside the control room. A dedicated operator will be stationed in the field so ANSI/ANS-58.8-1 Section 5.1 allows the use of performance data.

For the plant condition in which these actions will be needed is greater than 1 hour prior to core uncovery so the ANSI/ANS-58.8-1994 criteria is met. Calculation CN-OA-08-5 is the design basis calculation to validate that the core uncovery requirements are met. CN-OA-08-5 page 11 shows (Scenario #5) that greater than 2 hours to core uncovery exists for the proposed plant configuration. CN-OA-08-5 scenario #5 is for mid-loop conditions; the plant conditions when this action will be required are at greater inventory, lower temperature, and lower decay heat (due to new fuel) than the bounding analysis. So significantly more time is available prior to core uncovery. Calculation ECS10-002 provides the time to boil at reactor flange level and 23 days post trip as greater than 53 minutes. For the OP-903-115 performance, RCS inventory will be greater than flange level so additional time would be available. The 53 minutes prior to boiling is greater than the time required per ANSI/ANS-58.8-1994 to credit this OP-903-115 operator action to close SI-124A. No adverse impact.

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

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

BASIS:

This proposed change is not changing any equipment or processes. Therefore the proposed change does not create the possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the UFSAR.

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

   BASIS:

   The proposed change ensures the standby shutdown cooling train remains capable of performing it specified function. This ensures that the current limiting safety analysis for dose consequences and fission product barrier limits remain bounding. Therefore, the proposed change does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

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

   BASIS:

   The existing UFSAR evaluations pertaining to the shutdown cooling operations have not changed. Therefore, the proposed change does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses

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

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I. OVERVIEW / SIGNATURES[1]

Facility: Waterford 3 SES **Evaluation # / Rev. #:** 2011-01

Proposed Change / Document: Calculation HI-2104606, Rev 0, Missile Calculation Analysis for Waterford HI-STORM

Description of Change:
Dry Fuel Storage operations at WF3 will be conducted under the general license in accordance with Subpart K of 10 CFR Part 72. The ISFSI General License, issued by 10 CFR 72.210, "General License Issued," authorizes a 10 CFR Part 50 nuclear power plant licensee to store spent fuel at an onsite ISFSI in a storage system that is pre-approved by the NRC, provided that the licensee is in compliance with requirements of 10 CFR 72.212, "Conditions of General License Issued Under Part 72.210." 10 CFR 72.212(b)(6) requires that the WF3 reactor site parameters (seismic, tornado winds and missiles, ambient temperatures, etc.) be enveloped by the HI-STORM 100 Cask System design bases such that the HI-STORM 100 Cask System can safety withstand WF3 site specific environmental parameters.

Calculation HI-2104606, Rev. 0, Missile Calculation Analysis for Waterford HI-STORM, was issued for use at Waterford 3 by EC 14275. This calculation documents the evaluation of the Holtec HI-STORM dry fuel storage system when struck by tornado missiles defined in the Waterford 3 Updated Final Safety Analysis Report (UFSAR).

The missiles defined by the Holtec HI-STORM CFSAR, Table 2.2.5, are:

- Automobile – 1800 kg striking at 126 mph
- Rigid Solid Steel Cylinder – 8" Ø, 125 kg striking at 126 mph
- Steel Sphere – 1" Ø, 0.22 kg striking at 126 mph

The missiles defined by the Waterford 3 UFSAR, Table 3.5-10, are:

- Automobile – 4000 lb striking at 50 mph
- Utility Pole – 13.5" Ø, 1490 lb striking at 144 mph
- Schedule 40 Pipe – 3" Ø, 75.8 lb striking at 100 mph
- 2X4 Wood Plank – 27.8 lb striking at 300 mph
- Steel Rod – 1" Ø, 8 lb striking at 216 mph

The analysis determined that the automobile missile described in the WF3 UFSAR is bounded by the automobile missile analyzed in the HI-STORM CFSAR. However, the Holtec intermediate missile does not bound the three WF3 intermediate missiles or the one WF3 small missile based on comparing the impact energy of the different missiles. The calculation demonstrates that the design basis functions of the HI-STORM as described in the HI-STORM 100 CFSAR, Section 2.2.3.5, which are to maintain the kinematic stability of the HI-STORM overpack and continued integrity of the MPC confinement boundary while within the storage overpack, are satisfied with WF3 tornado generated missile impact on the HI-STORM. This calculation shows that the WF3 tornado generated missile impact will not cause the HI-STORM to overturn or suffer excessive sliding. The HI-STORM CFSAR section 11.2.3.1 conclusion that tornado missiles will not tip over the overpack is confirmed in this calculation.

The tornado missiles are not considered in the design of the ISFSI Pad since WF3 does not have an anchored HI-STORM 100 system. As described in the HI-STORM CFSAR, the only time the loads from a tornado missile need to be considered transmitted into the slab is when the overpack is anchored to the concrete. These anchored systems are only used in high seismic areas, while WF3 is in a relatively low seismic area and did not use the anchored HI-STORM system.

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

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The HI-STORM 100 CFSAR, Section 11.2.6.3, concludes that, while tornado missiles could cause localized damage to the radial shielding of a storage cask resulting in increased dose rates on contact with the affected area, the "...damage will have a negligible effect on the site boundary dose." The HI-STORM 100 CFSAR, Section 5.1.2 states "Design basis accidents which may affect the HI-STORM overpack can result in limited and localized damage to the outer shell and radial concrete shield. As the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose at the site boundary is negligible. Therefore, the site boundary, adjacent, and one meter doses for the loaded HI-STORM overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits."

References:
Calculation HI-2104606, Revision 0
Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System
EC 14275, Process Applicability Determination form

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

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

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

Preparer: Gregory N. Ferguson / Gregory N. Ferguson / EOI / DE-Civil / 10/17/11
Name (print) / Signature / Company / Department / Date

Reviewer: Jason Laque / Jason Laque / EOI / Maint / 10/17/11
Name (print) / Signature / Company / Department / Date

OSRC: Keith Nichols / Keith Nichols 10/19/11
Chairman's Name (print) / Signature / Date

W311-25
OSRC Meeting #

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II. 72.48 EVALUATION

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

☐ Yes ☒ No

Does the proposed Change:

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

   ☐ Yes ☒ No

   BASIS: THERE IS NO FREQUENCY OF OCCURRENCE SPECIFIED OR USED FOR TORNADO MISSILES. FOR ANY STRUCTURES REQUIRED TO RESIST TORNADO MISSILE DAMAGE, THE ASSUMPTION IS THAT A TORNADO MISSILE WILL OCCUR AND STRIKE THE EVALUATED TARGET. SINCE THE REQUIREMENT IS THAT WE ASSUME A TORNADO MISSILE WILL OCCUR, THERE CANNOT BE ANY INCREASE IN THE OCCURRENCE OF TORNADO MISSILES.

   CALCULATION HI-2104606 DOCUMENTS THAT THE IMPACT OF A TORNADO MISSILE AS DEFINED IN THE WF3 UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) WILL NOT CHANGE ANY RESULTS AS PREVIOUSLY EVALUATED IN THE CFSAR.

   THEREFORE, THE WF3 SPECIFIED TORNADO MISSILES ANALYZED IN CALCULATION HI-2104606 WILL NOT RESULT IN ANY INCREASE IN THE FREQUENCY OF OCCURRENCE OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE CFSAR.

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

   ☐ Yes ☒ No

   BASIS: CALCULATION HI-2104606 DETERMINED THE WF3 SPECIFIED TORNADO MISSILES WOULD CAUSE SOME LOCALIZED DAMAGE TO THE HI-STORM OVERPACK. THE HOLTEC HI-STORM 100 SYSTEM CONSISTS OF PASSIVE COMPONENTS WITH NO ACTIVE COMPONENTS.

   THE CALCULATION SHOWS THAT THE DAMAGE TO THE OVERPACK WILL NOT CAUSE ANY DAMAGE TO THE PASSIVE HEAT REMOVAL SYSTEM FOR THE MULTI-PURPOSE CANISTER (MPC) THAT CONTAINS THE SPENT NUCLEAR FUEL. THEREFORE, THE HEAT REMOVAL SYSTEM WILL CONTINUE TO FUNCTION PROPERLY.

   THEREFORE, THE WF3 SPECIFIED TORNADO MISSILES ANALYZED IN CALCULATION HI-2104606 WILL NOT RESULT IN ANY MALFUNCTION OF ANY SSC PREVIOUSLY EVALUATED IN THE CFSAR.

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

   ☐ Yes ☒ No

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BASIS: WHEN ANALYZING FOR TORNADO MISSILES, THE ASSUMPTION IS THAT A TORNADO MISSILE WILL OCCUR AND STRIKE THE EVALUATED TARGET, WHICH IN THIS CASE IS THE HOLTEC HI-STORM OVERPACK. CALCULATION HI-2104606 DOCUMENTS THAT THE IMPACT OF THE WF3 TORNADO MISSILES AS DEFINED IN THE UFSAR WILL NOT PREVENT THE HI-STORM SYSTEM FROM PERFORMING THE DESIGN FUNCTION OF SAFELY STORING THE SPENT NUCLEAR FUEL INSIDE THE MPC. THE POSSIBLE CONSEQUENCES OF A TORNADO MISSILE STRIKE ON THE HI-STORM OVERPACK ARE INCREASED RADIATION DOSE RATES AND LACK OF COOLING OF THE MPC.

CFSAR SECTION 5.1.2 DOCUMENTS THAT THE LOCALIZED DAMAGE FROM THE CFSAR ACCIDENTS WILL NOT INCREASE THE RADIATION DOSE RATES AT THE SITE BOUNDARY, ADJACENT TO THE ISFSI CONCRETE PAD, AND ONE METER FROM THE HI-STORM OVERPACK.

CALCULATION HI-2104606 DOCUMENTS THAT THE TORNADO MISSILES DEFINED IN THE WF3 UFSAR DO NOT RESULT IN ANY COMPROMISE OF THE INTEGRITY OF THE BOUNDARY OF THE HI-STORM SYSTEM. WHILE THE CALCULATION DOES SHOW THAT THERE WILL BE SOME LOCALIZED DAMAGE FROM THE TORNADO MISSILE STRIKE, THE DAMAGE WILL REMAIN WITHIN THE ALLOWED AMOUNTS AS DEFINED IN THE CFSAR AND WILL NOT ADVERSELY IMPACT THE COOLING CAPABILITY OF THE HI-STORM SYSTEM.

THE CALCULATION ALSO SHOWS THAT THE ENERGY FROM THE WF3 SPECIFIC TORNADO MISSILES WILL NOT CAUSE THE HI-STORM OVERPACK TO EITHER OVERTURN OR HAVE EXCESSIVE SLIDING. THESE RESULTS SHOW THAT THERE WILL NOT BE ANY CHANGE IN THE COOLING ABILITY OF THE HI-STORM SYSTEM.

THEREFORE, THE WF3 SPECIFIED TORNADO MISSILES ANALYZED IN CALCULATION HI-2104606 WILL NOT RESULT IN ANY INCREASE IN THE CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE CFSAR.

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

BASIS: THE HOLTEC HI-STORM 100 SYSTEM CONSISTS OF PASSIVE COMPONENTS, WITH NO ACTIVE COMPONENTS. THE HI-STORM CFSAR JUSTIFIES THAT EVEN THOUGH THERE WILL BE SOME LOCALIZED DAMAGE TO THE HI-STORM OVERPACK FROM THE CFSAR DEFINED TORNADO MISSILES, THE HI-STORM SYSTEM WILL STILL PROTECT THE SPENT NUCLEAR FUEL AND PROVIDE FOR PROPER COOLING. HI-2104606 ALSO SHOWS THAT THERE WILL BE LOCALIZED DAMAGE TO THE HI-STORM SYSTEM, BUT DOCUMENTS THAT THIS LOCALIZED DAMAGE WILL BE WITHIN THE ALLOWABLE LIMITS ALREADY DEFINED IN THE HI-STORM CFSAR.

THE CFSAR SECTION 5.1.2 DOCUMENTS THAT THE LOCALIZED DAMAGE FROM THE CFSAR DEFINED TORNADO MISSILES WILL NOT INCREASE THE RADIATION DOSE RATES AT THE SITE BOUNDARY, ADJACENT TO THE ISFSI CONCRETE PAD, AND ONE METER FROM THE HI-STORM OVERPACK. CALCULATION HI-2104606 SHOWS THAT THE LOCALIZED DAMAGE TO THE HI-STORM OVERPACK WILL REMAIN WITHIN ALLOWABLE LIMITS AS DOCUMENTED IN THE CFSAR, WHICH MEANS THE WF3 DEFINED TORNADO MISSILES WILL ALSO NOT CAUSE ANY INCREASED RADIATION DOSE RATES.

THEREFORE, THERE WILL NOT BE ANY INCREASE IN THE CONSEQUENCES OF A MALFUNCTION OF A SSC IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE CFSAR.

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

BASIS: CALCULATION HI-2104606 DOES NOT INTRODUCE ANY NEW OPERATING CHARACTERISTICS OR ACCIDENTS THAT NEED TO BE EVALUATED. THE CFSAR ALREADY INCLUDES ANALYSIS OF TORNADO MISSILES. THE CALCULATION IS USED TO ANALYZE THE IMPACT OF THE TORNADO MISSILES DEFINED IN THE WF3 UFSAR ON THE HOLTEC HI-STORM SYSTEM. ALL OF THE POTENTIAL CONCERNS FOR A TORNADO MISSILE STRIKE ARE INCLUDED IN THE HI-2104606 CALCULATION.

SINCE THIS CALCULATION IS NOT INTRODUCING ANY NEW SYSTEMS OR EQUIPMENT TO THE HI-STORM SYSTEM, AND DOES NOT INTRODUCE ANY OPERATING CHARACTERISTICS FOR THE HI-STORM SYSTEM, THERE WILL NOT BE ANY POSSIBILITY OF AN ACCIDENT OF A DIFFERENT TYPE THAN ANY PREVIOUSLY EVALUATED IN THE CFSAR.

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

BASIS: CALCULATION HI-2104606 DOES NOT INTRODUCE ANY NEW OPERATING CHARACTERISTICS OR EQUIPMENT TO THE HI-STORM SYSTEM. THE PURPOSE OF THIS CALCULATION IS TO ANALYZE THE IMPACT OF TORNADO MISSILES AS DEFINED IN THE WF3 UFSAR ON THE HI-STORM OVERPACK. THE HI-STORM CFSAR ALREADY INCLUDES TORNADO MISSILES AS AN ACCIDENT TO BE ANALYZED.

CALCULATION HI-2104606 SHOW THAT THE RESULTS OF A TORNADO MISSILE STRIKE ON THE HI-STORM OVERPACK WILL NOT BE ANY DIFFERENT THAN THAT ALREADY DOCUMENTED IN THE CFSAR. THERE WILL NOT BE ANY NEW MALFUNCTIONS INTRODUCED BY UTILIZING THIS CALCULATION. THEREFORE, THERE WILL NOT BE A POSSIBILITY FOR A MALFUNCTION OF A SSC IMPORTANT TO SAFETY WITH A DIFFERENT RESULT THAN ANY PREVIOUSLY EVALUATED IN THE CFSAR.

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

BASIS: THE OUTERMOST FISSION PRODUCT BARRIER FOR THE HI-STORM SYSTEM IS THE MPC. THE CFSAR SHOWS THAT THE TORNADO MISSILES DEFINED IN THE CFSAR WILL NOT CAUSE ANY DESIGN BASIS LIMITS FOR THE MPC OR THE SPENT FUEL INSIDE THE MPC TO BE EXCEEDED OR ALTERED.

THE ANALYSIS IN CALCULATION HI-2104606 DOCUMENTS THAT THE IMPACT OF TORNADO MISSILES AS DEFINED IN THE WF3 UFSAR WILL NOT CAUSE ANY DESIGN BASIS LIMITS FOR THE MPC OR THE SPENT FUEL INSIDE THE MPC TO BE EXCEEDED OR ALTERED.

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

BASIS: THE METHOD OF EVALUATION USED IN HI-2104606 IS THE CLASSIC ENERGY BALANCE METHODOLOGY, WHICH IS THE SAME METHODOLOGY USED IN THE CFSAR TO ANALYZE THE HI-STORM OVERPACK FOR THE CFSAR DEFINED TORNADO MISSILES.

THE ALLOWABLE LIMITS FOR THE OVERPACK DAMAGE FROM THE WF3 DEFINED TORNADO MISSILES ARE THE SAME AS THOSE ALREADY DEFINED IN THE CFSAR.

SINCE THE SAME DESIGN METHODOLOGY AND ALLOWABLE LIMITS ARE USED FOR THE TORNADO MISSILE ANALYSIS IN THE CFSAR AND IN HI-2104606, THERE IS NO DEPARTURE FROM A METHOD OF EVALUATION DESCRIBED IN THE CFSAR.

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