I. OVERVIEW / SIGNATURES

Facility: Waterford 3

Document Reviewed: ER-W3-2001-1149-005 **Change/Rev.:** 0

Description: EPU Impact on FSAR Chapter 5

System Designator(s)/Description: Various

Description of Proposed Change

ER-W3-2001-1149-005 is an Interdiscipline ER that provides the evaluation of the proposed Extended Power Uprate (EPU) impacts on FSAR Chapter 5 and the Systems, Structures and Components (SSCs) described in the FSAR Chapter. This ER provides input to ER-W3-2001-1149-000, the Nuclear Change ER that provides the justification for the acceptability of the implementation of EPU at Waterford 3 Steam Electric Station. ER-W3-2001-1149-005 revises FSAR Chapter 5 to address EPU-related impacts via DRN 03-2059. The DRN included with the ER package also includes editorial/typographical changes for the Chapter 5 text and tables identified during the review. The specific changes identified below satisfy the 50.59 Review Screening criteria as discussed in Section II of this review.

1. Table 5.1-1, "Process Data Point Tabulation" is revised to provide the full-power process conditions at the uprated operating point. The parameters are, temperature, mass flowrate, and volumetric flowrate.
2. Table 5.1-2 is revised to provide the full-power conditions at the uprated operating point. The parameters are, rated thermal power, thermal power, operating RCS flowrate, and RCS hot leg, cold leg, and average temperatures.
3. Section 5.2.2, Overpressurization Protection, subsection 5.2.2.1, is revised to identify the primary safety valves and main steam safety valves as providing the overpressurization protection.
4. Section 5.2.2, Overpressurization Protection, subsection 5.2.2.2 is revised to add a statement that the feedwater line break analysis of subsection 15.2.3 demonstrates that the pressurizer level remains below the primary safety valve inlet.
5. Appendix 5.2B, "Low Temperature Overpressure Protection During Heatup, Cooldown, and Cold Shutdown" is revised to delete Figures 5.2B-1, 5.2B-2, and 5.2B-3. The Figures provide transient pressure vs. time plots. The text in Section 5.2B.3 is revised to identify the peak pressure value and compare it to the limiting component design pressure.
6. Section 5.3.1.4, "Special Controls for Ferritic and Austenitic Stainless Steels" is revised to make clear that Regulatory Guide 1.99 was used without exception for the EPU analyses (the original analyses took exception to R.G. 1.99 on technical grounds).
7. Section 5.3.2.1, "Limit Curves" is revised to report the fluence and ART results for EPU conditions.
8. Section 5.3.2.3, "Fracture Toughness for Pressurized Thermal Shock Events" is revised to describe the analysis performed for EPU. Table 5.3-14 is added as a new Table.
9. Table 5.3-10, "Capsule Assembly Removal Schedule" is revised to provide the actual Removal Time (EFPY) for capsules already removed, to update the times for the remaining capsules, and to update the Lead Factors.
10. Section 5.4.1, Reactor Coolant Pumps, subsection 5.4.1.3 is revised to add a new paragraph:

    "The actual nominal RCS flowrate is approximately equal to 110% of the original design value of 99,000 gpm per pump. This results in a mass flowrate of approximately 165 Mlb/hr under normal operating conditions."
11. Section 5.4.1.4.1.2, "Flywheel Design Criteria," and section 5.4.1.4.2, "Additional Data and Analysis" are revised to note that the effects of the original design basis LOCA (i.e., main coolant loop pipe breaks, which have been eliminated via application of leak-before-break) envelope the effects of the next limiting set of pipe breaks (i.e., RCS branch line breaks).
12. Section 5.4.2, "Steam Generators" is revised to add the steam generator operating conditions at the new full power level, and to recast the original operating conditions into the past tense. Values are provided for

the following parameters: rated thermal power, NSSS thermal power, main steam flowrate, SG pressure, and feedwater temperature.

13. Section 5.4.14.2.3, "Reactor Coolant Pump Supports" is revised to note that original design basis LOCA (i.e., main coolant loop pipe breaks, which have been eliminated via application of leak-before-break) continue to provide enveloping design loads on the RCP supports.
14. Section 5.4.14.3, "Evaluation" is revised to replace the original description of the reactor vessel support analysis (and the commitment to provide additional analysis in a future FSAR amendment) with a description of the methodology used for the EPU analysis.
15. Table 5.4-1, "Reactor Coolant Pump Parameters" is revised to update the normal operating temperature to the EPU value.
16. Table 5.4-2, "Steam Generator Parameters" is revised to update the parameters to the EPU values.
17. Table 5.4-4, "Reactor Coolant Piping Parameters" is revised to update the normal operating RCP flowrate.
18. Figure 5.4-8, "Temperature Control Program" is revised to depict the RCS temperature vs. power relationship under EPU conditions.
19. Appendix 5.4A, "Dynamic Analysis for the Waterford 3 Reactor Vessel Support Loads Under LOCA Conditions" is revised to note that the appendix is retained for the historical record because the reactor vessel support loads, as well as RV shell response motions used in RV internals evaluations, that were generated from this analysis, bound the respective RV responses generated by the branch line pipe break analysis for EPU.

The ER package incorporates changes that satisfies the 50.59 Review Exemption criterion as discussed in Section III of this review:

1. FSAR Section 5.4.2, Steam Generators, incorporates a change in the moisture content of the steam exiting the moisture separator from 0.2% to 0.25%. This appears several times in the text of subsections 5.4.2.1 and 5.4.2.2, and in Table 5.4-2.

The ER package also incorporates editorial / typographical changes identified during the review. The editorial / typographical changes to FSAR Chapter 5 are:

1. Table 5.1-2, "Design Parameters of Reactor Coolant System" is renamed "Parameters of Reactor Coolant System" because it provides both design and normal operating values. The parameter descriptions are revised for clarity.
2. Section 5.2.5.1.2, Containment Airborne Particulate, Iodine and Gaseous Radioactivity Monitoring" is revised to indicate that the activity concentrations used to develop Table 5.2-10 results were based on 85 percent of the original thermal rating. Addition of the words "of the original" clarifies the basis of the Table. Table 5.2-10 summarizes the time to detect a 1 gpm RCS leak for the various leak detection devices, assuming 0.1% failed fuel. The original analysis is bounding for EPU since RCS activity is generally proportional to power.
3. There are two issues related to FSAR Appendix 5.2A, "Overpressure Protection for Combustion Engineering 3410 MWt Pressurizer Reactors". The first issue is that the information in the appendix is being removed from the FSAR; it is being replaced by an equivalent licensee-controlled engineering calculation (DAR-OA-03-10). The calculation is being incorporated by reference into FSAR Appendix 5.2A. The relocation of the Overpressure Protection analysis to a licensee-controlled engineering calculation incorporated by reference falls under the category of "FSAR Only" change as described in LI-113. The second issue is the evaluation of the revised analysis which is discussed in Section IV, 50.59 Evaluation.
4. Appendix 5.2B, "Low Temperature Overpressure Protection During Heatup, Cooldown, and Cold Shutdown" is revised to reflect the LTOP relief valve sizing calc done for EPU.

   Overpressure protection of the Reactor Coolant System (RCS) during low temperature conditions is provided by relief valves located in the Shutdown Cooling System (SDCS) suction line. This protection precludes any overpressurizing transient from exceeding the pressure-temperature (P-T) operating limits provided in the Technical Specifications. The protection provided by these relief valves is required during heat up and cooldown and during extended periods of cold shutdowns. The SDCS relief valve is a spring-loaded (bellows) liquid relief valve with sufficient capacity to mitigate the most limiting overpressurization event. The analysis of the LTOP transients is described in FSAR Appendix 5.2B, "Low Temperature Overpressure Protection During Heatup, Cooldown, and Cold Shutdown."

The limiting transients for the LTOP relief valves are (1) a mass addition transient (inadvertent actuation of safety injection) and (2) an energy addition transient (starting an RCP when the steam generators are at a higher temperature than the RCS). The original valve capacity requirement was identified by CE as 3089 gpm per valve, in interface requirement letter C-CE-2429 (1975). Subsequent analysis documented in C-PEC-117 (1976) identified a required capacity of 3004 gpm. The analyses which support the LTOP discussion in FSAR Appendix 5.2B, including FSAR Figure 5.2B-3, "Limiting Transients With SDC Relief Valve Protection," are documented in C-PEC-185 and C-PEC-187 (1978). These analyses assume a capacity of 3089 gpm at 10% accumulation. The evaluation for power uprate to 3716, documented in CN-PS-03-10 (2003) also assumes a capacity of 3089 gpm at 10% accumulation.

Pertinent valve parameters and assumptions used in the analyses are provided in Section 5.2B.3.c of Appendix 5.2B One of the parameters listed is the relief valve capacity, which is currently identified as 3004 gpm. .This is inconsistent with the 3089 gpm value used in the pre-EPU analyses C-PEC-185 and C-PEC-187 described above.

The FSAR stated capacity of 3004 gpm is being revised to 3089 gpm at 10% accumulation to be consistent with the original analyses (as presented in the FSAR currently via Figure 5.2B-3). This value is also consistent with the analyses done in support of the power uprate to 3716 MWt.

The capacity of the installed valves is 3345 gpm, which exceeds the assumed capacity of 3089 (both at 10% accumulation). The EPU analysis indicates that the required capacity is below 2000 gpm (i.e., the valve does not achieve 10% accumulation in the limiting transient).

5. Section 5.2.2, Overpressurization Protection, subsection 5.2.2.2 is revised to change the citation to subsection 15.2.1.1 (loss of external load) to 15.2.1.3 (loss of condenser vacuum). The description in 15.2.1.1 states that the sequence of events for the loss of external load event would produce consequences no more adverse than those following a loss of condenser vacuum, which is described in Subsection 15.2.1.3. It is therefore considered more appropriate to cite 15.2.1.3 directly.
6. Section 5.3.1.4, "Special Controls for Ferritic and Austenitic Stainless Steels" is revised to make format changes (to align paragraphs under their heading properly) and to change "Combustion Engineering" to "Westinghouse" as appropriate.
7. Section 5.3.1.6.1, "Test Materials Selection" corrects a typographic error ("tower shells" should be "lower shells").
8. Section 5.3, "References" is revised to add five new references. This change falls under the category of "FSAR Only Change" as defined in LI-113.
9. Section 5.4.14, "Components Support" is revised to use the term "pipe break" in place of "LOCA" for clarity. Several other minor editorial changes are made including reference to subsection 3.6.3, which is a new section added via ER-W3-2001-1149-003, which discusses LBB..
10. Table 5.4-1, "Reactor Coolant Pump Parameters" is revised to clarify the actual meaning of the given RCP volumetric flowrate. The 99,000 gpm is the "rated flow" for the pump; the term "design flow" is ambiguous since some analyses use minimum flow rate, while others use maximum, and still others use nominal values.

No further evaluation of these editorial / typographical changes is required by this 50.59 review.

The ER package incorporates the following change that requires a 50.59 Evaluation as discussed in Section IV of this review:

There are two issues related to FSAR Appendix 5.2A, "Overpressure Protection for Combustion Engineering 3410 MWt Pressurizer Reactors". The first issue is that the information in the appendix is being removed from the FSAR; it is being replaced by an equivalent licensee-controlled engineering calculation (DAR-OA-03-10). The calculation is being incorporated by reference into FSAR Appendix 5.2A. The relocation of the Overpressure Protection analysis to a licensee-controlled engineering calculation incorporated by reference falls under the category of "FSAR Only" change as described in LI-113. The second issue is the evaluation of the revised analysis which is discussed in Section IV, 50.59 Evaluation.

Calculations were completed in support of the evaluation of the SSCs, analyses and evaluations discussed in Chapter 5 of the FSAR. Each of these calculations and analyses is summarized in ER-W3-2001-1149-005 with regards to the purpose of the calculation and the conclusions reached in the calculation. Applicable criteria specified in the calculations, such as ASME Code limits, are met, and the SSCs evaluated are determined to be acceptable for operation at EPU conditions. No hardware changes are required as a result of these calculations and analyses done in support of the FSAR Chapter 5 SSC evaluations.

For the Chapter 5 SSCs, the analyses include determination of the new operating point (including the thermal hydraulic evaluation of the steam generators), and the changes to the component specifications resulting from operation at the new point. Changes to the specifications drive the reanalysis of the components. Portions of the revised structural analyses are encompassed in this ER, other portions are encompassed by the Chapter 3 ER. In addition to the structural analyses, system-level analyses have been performed to ensure that the RCS continues to meet its design basis requirements.

Check the applicable review(s): (Only the sections indicated must be included in the Review.)

☒	EDITORIAL CHANGE of a Licensing Basis Document	Section I
☒	SCREENING	Sections I and II required
☒	50.59 EVALUATION EXEMPTION	Sections I, II, and III required
☒	50.59 EVALUATION (#: 04-008)	Sections I, II, and IV required

Preparer: W.H. Chenault / [signature] / Enercon / DE / 12/2/04
Name (print) / Signature / Company / Department / Date

Reviewer: Ralph Schwartzbeck / R.S. Nobles for R. Schwartzbeck per Telecon / Enercon / DE / 12/2/04
Name (print) / Signature / Company / Department / Date

OSRC: R. A. Dodds III [signature] 12/2/4
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

II. SCREENINGS

A. Licensing Basis Document Review

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

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113. (See Section 5.2[13] for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	ER-W3-2001-1149-005, DRN No. 03-2059
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Report and supplements for the initial FSAR[1]	☐	☒	
NRC Safety Evaluations for amendments to the Operating License[1]	☐	☒	
If "YES", perform an Exemption Review per Section III <u>OR</u> perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change. If obtaining NRC approval, document the LBD change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC. <u>AND</u> initiate an LBD change in accordance with NMM LI-113.			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2, 3]	☐	☒	
Fire Protection Program[3, 4] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual[3, 4]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation <u>AND</u> initiate an LBD change in accordance with NMM LI-113. No further 50.59 review is required.			

[1] If "YES," see Section 5.2[5]. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Review.
[3] Changes to the Emergency Plan, Fire Protection Program, and Offsite Dose Calculation Manual must be approved by the OSRC in accordance with NMM OM-119.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition or under 50.59, as appropriate.

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No

If "yes," perform a 50.59 Evaluation per Section IV <u>OR</u> obtain NRC approval prior to implementing the change <u>AND</u> initiate an LBD change in accordance with NMM LI-113. If obtaining NRC approval, document the change in Section II.A.5; no further 50.59 review is required. However, the change cannot be implemented until approved by the NRC.

3. **<u>Basis</u>**

Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Discuss other LBDs if impacted. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.3.2 for guidance.

ER-W3-2001-1149-005 is an Interdiscipline ER with the scope of providing the evaluation of FSAR Chapter 5 and any required FSAR changes as a result of the proposed implementation of EPU. The ER documents evaluation of the Waterford 3 Extended Power Uprate (EPU) effects on the FSAR Chapter 5 SSCs, and associated design and licensing documentation. Operation under EPU conditions may require plant SSCs to operate under different conditions, meet different performance criteria, or perform new or different functions. The SSCs described in Chapter 5 of the FSAR perform critical safety functions as well as functions important to power generation. Evaluation of the FSAR Chapter is necessary to address the potential for EPU to affect these SSCs and their functional requirements. While the ER will identify the effects of EPU on these SSCs and associated documentation, other ERs will evaluate the affected plant systems and implement any required changes to ensure readiness for operation under EPU.

Operating License:

The Waterford Unit 3 operating license is impacted by the Extended Power Uprate; however, license amendment request NPF-38-249 for EPU (including all supplements) addresses those applicable changes. The current Operating License/Technical Specifications identifies the licensed thermal power level as 3441 MWt. License amendment request NPF-38-249 addresses the Waterford 3 request to implement an Extended Power Uprate (EPU) that will increase the licensed thermal power level to 3716 MWt. This change requires NRC approval and is further documented in Section II.A.5. The operating license does not have any restrictions on activities within the scope of this change. None of the license conditions contained in the operating license are impacted by the activity within the scope of this ER beyond those addressed in the license amendment request; therefore, the proposed changes do not impact the Waterford Unit 3 operating license.

Technical Specifications:

The Waterford Unit 3 Technical Specifications are impacted by the Extended Power Uprate; however, license amendment request NPF-38-249 for EPU addresses those applicable changes. Implementation of EPU is dependent upon NRC acceptance of the Waterford 3 operating license via License Amendment Request NPF-38-249. This ER is an integral part of EPU; therefore, NRC approval of the license amendment request is required for implementation of this ER's activities.

Technical Specification changes included in LAR NPF-38-249 that are associated with changes made by this ER are as follows:

- <u>Reactor Coolant Cold Leg Temperature, Specification 3.2.6</u>

  The cold leg temperature range is changed from 541 °F – 558 °F to 536 °F - 549 °F to maintain margin to the new operating point nominal temperature of 543 °F at EPU conditions. Safety and structural analyses were performed consistent with the revised temperature range, and results presented in the PUR demonstrate acceptable results. In addition, the temperature of 568 °F that is allowed for 30 minutes after a power cutback was changed to 559 °F to maintain the existing 10 °F difference to maximum T_{cold}.

- Pressurizer Pressure, Specification 3.2.8

  The minimum pressurizer pressure value of 2025 psia is increased to 2125 psia to regain thermal margin under EPU conditions. Safety analyses were performed consistent with this proposed value and demonstrate acceptable results.

- Operational Leakage, Specification 3.4.5.2 c

  The steam generator primary to secondary leakage rate of 1 gpm through all steam generators, and 720 gallons per day through any one generator is replaced by a limit of 75 gallons per day through any one generator. This leak rate reduction is made to provide margin for use in dose consequence analysis at EPU conditions. Dose consequences for events in which steam generator leakage is relevant demonstrate acceptable results.

- Pressurizer Heatup/Cooldown, Specification 3/4.4.8.2

  The following are deleted from Specification 3/4.4.8.2:

  - Requirement 3.4.8.2c regarding the maximum spray nozzle usage factor of 0.65.
  - Action statement b regarding requirements if the spray nozzle usage factor exceeds 0.65 and reference to Table 5.7-1.
  - Surveillance requirement 4.4.8.2.2 regarding determining every 12 hours that spray water differential temperature is within limits during auxiliary spray operation.
  - Surveillance requirement 4.4.8.2.3 regarding recording of spray cycles and corresponding water differential temperature when spray is initiated with DT greater than 130 °F and when auxiliary spray is initiated with DT greater than 140 °F.

  These deletions are being made because the requirements to which they pertain in Technical Specification Table 5.7-1 are being deleted.

- Section 5.7, Component Cyclic or Transient Limits, and Table 5.7-1

  Transient logging requirements for the pressurizer spray valves, contained on the second and third pages of Table 5.7-1, are deleted based on calculations showing that transient logging is not required. The deleted provisions of Technical Specification 3/4.4.8.2 identified above pertain to these deleted transient logging requirements.

  Evaluations were performed in response to NRC Bulletin 88-08 of systems connected to the RCS that may be subjected to thermal stratification. These evaluations included defining new transients for the pressurizer spray system based on Waterford 3 plant-specific conditions. The new transients were defined to account for potential thermal shocks due to initiation of main and auxiliary pressurizer spray. The effects of reactor coolant pump operation on main spray bypass flow were also considered because characteristics of the bypass flow also affect the spray nozzle.

  The existing stress and fatigue analysis of the spray nozzle was revised. The revised analysis included evaluation of the new transients. In addition, the revised analysis included an inelastic analysis of the nozzle and a more realistic assessment of the fatigue calculation for the original design basis transients. For example, the original analysis used a carbon steel fatigue curve for the safe end region of the spray nozzle, a conservative approach for the nozzle material. Consequently, an appropriate stainless steel fatigue curve was used in the reanalysis. The analysis was performed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition with Addenda through Summer, 1971. Based on the more realistic evaluation of the original spray system transients and the new plant-specific Waterford 3 transients, the spray nozzle reanalysis showed that fatigue of the nozzle will not be significant over the 40 year plant design life. Consequently, the spray nozzle requirements in Table 5.7-1 that pertain to transient logging and calculation of the spray nozzle usage factor may be deleted.

  Section 5.7 and the remainder of Table 5.7-1 are being relocated to the TRM in order to achieve a level of detail consistent with NUREG-1432.

The changes within the scope of this ER do not require revision to the Technical Specifications beyond those included in the EPU LAR as addressed above. The activities within the scope of this ER will not adversely affect the mode of operation of any important to safety equipment or Technical Specification associated equipment. In addition, the activities will not create a system configuration or operating condition such that a Technical Specification LCO or surveillance requirement is no longer adequate. Likewise, the activities will not result in a condition that would bypass or invalidate automatic actuation features required to be operable by the Technical Specifications or exceed any limits specified in the Operating License and Technical Specifications. Therefore, the proposed changes do not require an Operating License or Technical Specification change that is not included in LAR NPF-38-249.

FSAR:

1. Table 5.1-1, "Process Data Point Tabulation" is revised to provide the full-power process conditions at the uprated operating point. The parameters are, temperature, mass flowrate, and volumetric flowrate. The revised values are consistent with the uprate operating point information discussed in LAR-NPF-38-249 (see for example, PUR Table 1-2).
2. Table 5.1-2 is revised to provide the full-power conditions at the uprated operating point. The parameters are, rated thermal power, thermal power, operating RCS flowrate, and RCS hot leg, cold leg, and average temperatures. The revised values are consistent with the uprate operating point information discussed in LAR-NPF-38-249, PUR Tables 1-2 and 2.2-8.
3. Section 5.2.2, Overpressurization Protection, subsection 5.2.2.1, is revised to identify the primary safety valves and main steam safety valves as providing the overpressurization protection. This is consistent with LAR-NPF-38-249 (PUR Section 2.6.4.2).
4. Section 5.2.2, Overpressurization Protection, subsection 5.2.2.2 is revised to add a statement that the feedwater line break analysis of subsection 15.2.3 demonstrates that the pressurizer level remains below the primary safety valve inlet. This is consistent with LAR-NPF-38-249 (PUR Section 2.13.2.3.1), Where it states that operator action to secure charging is credited before 12 minutes to avoid filling the pressurizer.
5. Appendix 5.2B, "Low Temperature Overpressure Protection During Heatup, Cooldown, and Cold Shutdown" is revised to delete Figures 5.2B-1, 5.2B-2, and 5.2B-3. The Figures provide transient pressure vs. time plots. They are deleted because the only important result is that the peak pressure remains within the specified limits – the time of the peak is completely immaterial. The text in Section 5.2B.3 is revised to identify the peak pressure value and compare it to the limiting component design pressure (i.e., SDC component, LPSI pump seal). This is consistent with the information discussed in LAR-NPF-38-249 (PUR Section 2.6.4.3), which stated that the pressure limits are not exceeded (though it did not provide the quantitative results).
6. Section 5.3.1.4, "Special Controls for Ferritic and Austenitic Stainless Steels" is revised to make clear that Regulatory Guide 1.99 was used without exception for the EPU analyses (the original analyses took exception to R.G. 1.99 on technical grounds). The use of R.G. 1.99 in the P-T and vessel beltline materials analyses is discussed in LAR-NPF-38-249, PUR Section 2.1.2.
7. Section 5.3.2.1, "Limit Curves" is revised to report the fluence and ART results for EPU conditions. The revised values are consistent with the values discussed in LAR-NPF-38-249 (PUR Section 2.1.1, Tables 2.1-1 and 2.1-2) and in the documents associated with License Amendment 196 (see ER-W3-2004-0439). Note that the 50°F ART value has been submitted and approved via WCAP 16088 and the P-T limits TS Amend 196.
8. Section 5.3.2.3, "Fracture Toughness for Pressurized Thermal Shock Events" is revised to describe the analysis performed for EPU. Table 5.3-14 is added as a new Table; the table is based on LAR-NPF-38-249 PUR Table 2.1-2. The revised discussion is consistent with the discussion in LAR-NPF-38-249 (though it provides more detailed information) PUR Sections 2.1.2 and 2.1.3 and Table 2.1-1.
9. Table 5.3-10, "Capsule Assembly Removal Schedule" is revised to provide the actual Removal Time (EFPY) for capsules already removed, to update the times for the remaining capsules, and to update the Lead Factors. These changes are consistent with the discussion in LAR-NPF-38-249 (PUR Sections 2.1.1, 2.1.2, and 2.1.3), and WCAP-16002-NP, Analysis of Capsule 263 from the Entergy Operations Waterford Unit 3 Reactor Vessel Radiation Surveillance Program, Revision 0, March 2003. (Referenced in the LAR, Submitted to the NRC by Entergy letter W3F1-2003-0020, dated March 28, 2003, and associated with License Amendment 196 - see ER-W3-2004-0439.)
10. Section 5.4.1, Reactor Coolant Pumps, subsection 5.4.1.3 is revised to add a new paragraph:

"The actual nominal RCS flowrate is approximately equal to 110% of the original design value of 99,000 gpm per pump. This results in a mass flowrate of approximately 165 Mlb/hr under normal operating conditions."

This is added to clarify the basis for the nominal flow value used in various places throughout the text. The 110% value is mentioned in LAR-NPF-38-249 (Section 1.2) and is discussed in Attachment 2 to Entergy letter W3F1-2004-0037; the 99,000 gpm design value is already provided in the FSAR (for example, in Table 5.4-1). The 165 Mlb/hr value is consistent (to three significant digits) with the 45,808 lbm/sec value provided in LAR-NPF-38-249 (PUR Table 1-2).

11. Section 5.4.1.4.1.2, "Flywheel Design Criteria," and Section 5.4.1.4.2, "Additional Data and Analysis" are revised to note that the effects of the original design basis LOCA (i.e., main coolant loop pipe breaks, which have been eliminated via application of leak-before-break) envelope the effects of the next limiting set of pipe breaks (i.e., RCS branch line breaks). The LBB methodology was previously approved by the NRC in CEN-367-A and the acceptability for use of this methodology at Waterford in support of EPU is documented in CIN-2002- 00266 and discussed in LAR NPF-38-249. Further, the LAR states that the specified loads on the RCP motor are unchanged for EPU (Section 2.2.2.1.4.4.4). The flywheel is evaluated in DAR-CI-03-25, which determines that the original loads remain bounding for EPU.

12. Section 5.4.2, "Steam Generators" is revised to add the steam generator operating conditions at the new full power level, and to recast the original operating conditions into the past tense. Values are provided for the following parameters: rated thermal power, NSSS thermal power, main steam flowrate, SG pressure, and feedwater temperature. The updated parameter values are consistent with the uprate operating point and SG thermal-hydraulic performance information discussed in LAR-NPF-38-249 (PUR Table 2.2-8).

13. Section 5.4.11, "Quench Tank" is revised to delete the discussion of a loss-of-load event followed by a rod withdrawal incident. This is replaced by a statement that the tank sizing considers the bounding loss-of-load events. This change in the design basis event (eliminating the need to compound the loss-of-load event and the rod withdrawal incident) is discussed in LAR-NPF-38-249 (Section 2.5.2.7). The Analysis is documented in DAR-PS-03-2, Rev 1. Even though this FSAR change is within the scope of ER-W3-2001-1149-005, the 50.59 review is being performed within the scope of ER-W3-2001-1149-000 (see Section II.A.5 of this 50.59 Review).

14. Section 5.4.14.2.3, "Reactor Coolant Pump Supports" is revised to note that original design basis LOCA (i.e., main coolant loop pipe breaks, which have been eliminated via application of leak-before-break) continue to provide enveloping design loads on the RCP supports. The LBB methodology was previously approved by the NRC in CEN-367-A and the acceptability for use of this methodology at Waterford in support of EPU is documented in CIN-2002- 00266 and discussed in LAR NPF-38-249 (for the RCP supports, see specifically Section 2.2.2.1.4.4.5).

15. Section 5.4.14.3, "Evaluation" is revised to replace the original description of the reactor vessel support analysis (and the commitment to provide additional analysis in a future FSAR amendment) with a description of the methodology used for the EPU analysis. The description of the methodology is consistent with the description provided in LAR NPF-38-249 (Section 2.2.1.2).

16. Table 5.4-1, "Reactor Coolant Pump Parameters" is revised to update the normal operating temperature to the EPU value of 543 °F. The updated parameter value is consistent with the uprate operating point information discussed in LAR-NPF-38-249 (see for example, Table 1-2).

17. Table 5.4-2, "Steam Generator Parameters" is revised to update the parameters to the EPU values. The updated parameter values are consistent with the uprate operating point information discussed in LAR-NPF-38-249 (Table 1-2, Section 2.2.2.1.4.6.1, and Table 2.2-8). The overall heat transfer coefficient is not discussed in the LAR. The value in the Table is revised to one more consistent with recent analyses, and identified as "nominal"; heat transfer coefficient is not discussed in the text of the FSAR and is in the table for information only. The blowdown flow parameter is changed from "maximum" to "nominal" and the value changed to 165 gpm, consistent with the discussion in Section 2.1.10 of the LAR.

18. Table 5.4-4, "Reactor Coolant Piping Parameters" is revised to update the normal operating RCP mass flowrate. The updated parameter value is consistent with the uprate operating point information

discussed in LAR-NPF-38-249 (see for example, Table 1-2).

19. Figure 5.4-8, "Temperature Control Program" is revised to depict the RCS temperature vs. power relationship under EPU conditions. The revised temperature program is consistent with the uprate operating point information discussed in LAR-NPF-38-249 (Section 1.2, Section 2.13.0.2, Table 1-1 and Table 1-2).

20. Appendix 5.4A, "Dynamic Analysis for the Waterford 3 Reactor Vessel Support Loads Under LOCA Conditions" is revised to note that the appendix is retained for the historical record because the reactor vessel support loads, as well as RV shell response motions used in RV internals evaluations, that were generated from this analysis, bound the respective RV responses generated by the branch line pipe break analysis for EPU. See LAR NPF-38-249 (Sections 2.2.1.2 and 2.2.2.1.4.1.4).

Technical Specification Bases:

The Waterford Unit 3 Technical Specifications Bases are impacted by the Extended Power Uprate. However, EPU-related changes to the Technical Specifications are included in DRN 04-1243 and are evaluated separately by ER-W3-2001-1149-000. The bases for Tech Specs 3.2.6, 3.2.8, 3/4.4.5.2, and 3/4.4.8.2 were revised as discussed in Attachment 3 to Entergy Letter W3F1-2003-0074, which transmitted the EPU LAR to the NRC. These bases changes are consistent with the technical specification changes discussed in the Technical Specification section above.

Technical Requirements Manual (TRM):

The Waterford Unit 3 Technical Requirements Manual is impacted by the Extended Power Uprate. However, EPU-related changes to the Technical Requirements Manual are included in DRN 04-1244 and are evaluated separately by ER-W3-2001-1149-000.

TRM changes pertinent to FSAR Chapter 5 are as follows.

- TS Table 5.7-1, Component Cyclic or Transient Limits is being modified to remove the limits on the pressurizer spray nozzle transients, and the remainder of the table is being relocated to the TRM.
- A new TRM (section 3.4.3.1) is being added to require that the pressurizer heaters shall be OPERABLE with at least 650 kW of nominal heater capacity available in addition to the heater capacity specified in Technical Specification 3.4.3.1 b. This capacity is sufficient, in conjunction with the heater capacity required by Technical Specification 3.4.3.1 b, to bound the capacity credited in the analysis of the CEA Withdrawal within Deadband event. The additional heater capacity cited in requirement 3.4.3.1 can be heaters powered from any combination of Class 1E or non-class 1E.

NRC Safety Evaluation Report and supplements for the initial FSAR:

See Section IV, Evaluation, for a discussion of the changes to the Overpressure Protection analysis (FSAR Appendix 5.2A) that is being revised by this ER. This subject was evaluated in the SER and the evaluation determines any impacts to the information reviewed by the NRC.

New Test or Experiment:

This activity does not involve a test or experiment that is not described in the FSAR. The activities are restricted to evaluations of FSAR Chapter 5 SSCs with regard to EPU impacts and associated document changes. There are no tests or experiments included within the scope of this ER.

Other Impacts:

No other LBDs are impacted by the changes proposed in ER-W3-2001-1149-005.

4. <u>References</u>

Discuss the methodology for performing LBD searches. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.4.1[5]](d) of LI-101. **NOTE: Ensure that manual searches are performed using controlled copies of the documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
The LBD listed in Section II.A.1	Primary safety valve, secondary safety valve, pressurizer safety valve

LBDs/Documents reviewed manually:

FSAR Chapter 5

5. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10 CFR 50.59 Program Review Guidelines.)

☒ Yes

☐ No

If "YES", list the required changes/submittals. The changes covered by this 50.59 Review 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.

Although this ER does not initiate any Operating License or Technical Specification changes, some of the changes within the scope of this ER are dependent on changes included in the Extended Power Uprate License Amendment Request NPF-38-249 (including all supplements). The license amendment request is currently awaiting NRC approval and the approved NRC Safety Evaluation Report addressing this request must be reviewed to ensure that the assumptions made in this evaluation remain valid. An ERD action has been created for ER-W3-2001-1149-000 to review this ER against the approved/issued NRC SER to ensure that the ER remains in agreement with the approved SER.

FSAR Section 5.4.11, "Quench Tank" is revised by this ER to delete the discussion of a loss-of-load event followed by a rod withdrawal incident. This is replaced by a statement that the tank sizing considers the bounding loss-of-load events. The 50.59 Review for this change is within the scope of ER-W3-2001-1149-000. An ERD action (RFT33) has been created for ER-W3-2001-1149-000 to perform this evaluation.

B. ENVIRONMENTAL SCREENING

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

Will the proposed Change being evaluated:

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

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

C. SECURITY PLAN SCREENING

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

Could the proposed activity being evaluated:

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

Documentation for accepting any "yes" statement for these reviews will be attached to this 50.59 Review or referenced below.

D. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) SCREENING

If any of the following questions is answered "yes," an ISFSI Review must be performed in accordance with NMM Procedure LI-112, "72.48 Review," and attached to this Review.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Any activity that directly impacts spent fuel cask storage or loading operations?
2.	☐	☒	Involve the Independent Spent Fuel Storage Installation (ISFSI) including the concrete pad, security fence, and lighting?
3.	☐	☒	Involve a change to the on-site transport equipment or path from the Fuel Building to the ISFSI?
4.	☐	☒	Involve a change to the design or operation of the Fuel Building fuel bridge including setpoints and limit switches?
5.	☐	☒	Involve a change to the Fuel Building or Control Room(s) radiation monitoring?
6.	☐	☒	Involve a change to the Fuel Building pools including pool levels, cask pool gates, cooling water sources, and water chemistry?
7.	☐	☒	Involve a change to the Fuel Building handling equipment (e.g., bridges and cask cranes, structures, load paths, lighting, auxiliary services, etc)?
8.	☐	☒	Involve a change to the Fuel Building electrical power?
9.	☐	☒	Involve a change to the Fuel Building ventilation?
10.	☐	☒	Involve a change to the ISFSI security?
11.	☐	☒	Involve a change to off-site radiological release projections from non-ISFSI sources?
12.	☐	☒	Involve a change to spent fuel characteristics?
13.	☐	☒	Redefine/change heavy load pathways?
14.	☐	☒	Fire and explosion protection near or in the on-site transport paths or near the ISFSI?
15.	☐	☒	Involve a change to the loading bay or supporting components?
16.	☐	☒	New structures near the ISFSI?
17.	☐	☒	Modifications to any plant systems that support dry fuel storage activities?
18.	☐	☒	Involve a change to the nitrogen supply, service air, demineralized water or borated water system in the Fuel Building?

III. 50.59 EVALUATION EXEMPTION

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

A. Check the applicable boxes below. If any of the boxes are checked, clearly document the basis in Section II.B, below. If none of the boxes are appropriate, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.

☑ The proposed activity meets all of the following criteria regarding design function per Section 5.5[1](a):

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

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

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

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.5[1](b). Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.5[1](c). Reference: ____________________

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10 CFR 50.59 Review Program Guidelines for guidance.

Steam Moisture Content

FSAR Section 5.4.2: This FSAR Section provides a description of the steam generators. FSAR Section 5.4.2.1, 5.4.2.2, and Table 5.4-2 currently state that steam generated in the shell side of the steam generator flows through moisture separators that reduce moisture content to less than 0.2%. The EPU-related change will revise the moisture content of the steam exiting the moisture separator from 0.2% to 0.25%. FSAR Section 1.2.2.1.2 also refers to a moisture content of 0.2%; ER-W3-2001-1149-002 will revised this section of the FSAR to reflect a moisture content of 0.25% and will be evaluated by the 50.59 review associated with that ER package.

Westinghouse specification 9270-PE-120, Project Specification for Steam Generator Assemblies for Waterford Unit 3, provides the moisture content performance value. The specification has been revised to reflect EPU operating conditions and changes the maximum moisture content from 0.20% to 0.25%.

Calculation CN-SGDA-03-25, Thermal Hydraulic Analysis of Waterford-3 Steam Generators at 3716 MWt Power Uprate Conditions, has been generated for EPU and determines the analytical maximum moisture content of 0.22% for EPU operating conditions. This value is identified in PUR Table 2.2-9 along with the current analytical maximum moisture content of 0.147%. The impact review performed for this calc included electronic searches using "moisture carryover" and "circulation ratio."

The design function of the moisture separator is to reduce the amount of moisture carryover from the steam exiting the steam generator. The moisture separator will continue to function to reduce moisture carryover; however, the quantity and conditions of the steam change as a result of EPU conditions resulting in a change to the moisture content of the steam exiting the moisture separator. EOI specification DES-M-16, High Pressure Turbine Steam Path Replacement, has been developed for EPU and identifies that the design of the main turbine assumes a moisture content of the steam at the SG outlet as 0.38%. The revised SG specification moisture content of the steam remains well below this value.

Steam generator moisture carryover is an input to the secondary calorimetric power measurement as determined by COLSS. The secondary calorimetric power measurement uncertainty analysis assumes a main steam moisture carryover range of 0.0 to 0.4%. A change in the steam generator specified performance for maximum steam carryover to 0.25% does not affect uncertainty analysis or secondary calorimetric power measurement.

The slight increase in the SG outlet moisture carryover will not have an adverse impact on the main steam line steam traps and drains, since these are sized to accommodate the heavy condensation rates experienced during plant heatup when the lines are warmed.

The expected increase in the moisture carryover under EPU conditions has been included in the Flow-Accelerated Corrosion (FAC) CHECWORKS Model.

The FSAR will be revised to reflect the change in the specified performance of the moisture separator that correlates with the analytical value consistent with the current discussion provided in the FSAR; therefore, this proposed activity does not result in any adverse impacts on the design function of the SSC as described in the FSAR, does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR, and does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

IV. 50.59 EVALUATION

License Amendment Determination

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

☐ Yes
☒ No

Does the proposed Change:

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

   ☐ Yes
   ☒ No

   BASIS:

   Regarding changes to FSAR Appendix 5.2A

   FSAR Section 15.6.3.4, Inadvertent Opening of a Pressurizer Safety Valve, evaluates an accident initiated by a pressurizer safety valve. FSAR Section 15.6.3.4.1 states: "The inadvertent opening of a pressurizer safety valve at normal RCS operating pressures could only be caused by a passive mechanical failure of the valve." The changes in the valve analysis to determine the adequacy of the capacity of the pressurizer safety valves does not affect the capabilities of the valve, the reliability of the valves or change the function of the valves, it only determines if the valves are still adequate to perform their design basis function following EPU. Therefore, because the reliability and function of the valves have not been changed, the frequency of occurrence of this accident evaluated in the FSAR has not been increased.

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

   ☐ Yes
   ☒ No

   BASIS:

   Regarding changes to FSAR Appendix 5.2A

   As stated in Question 1 above, FSAR Section 15.6.3.4, Inadvertent Opening of a Pressurizer Safety Valve, evaluates an accident initiated by a malfunction of a pressurizer safety valve. FSAR Section 15.6.3.4.1 states: "The inadvertent opening of a pressurizer safety valve at normal RCS operating pressures could only be caused by a passive mechanical failure of the valve." The revision to the Overpressure Report does not change the function, reliability, setpoints, setpoint tolerance, or capacity of the primary or secondary safety valves, it only verifies that the valves are adequate to perform there design basis functions, i.e., meet the ASME code requirements, following EPU. Therefore, because the reliability and function of the valves have not been changed and the valves continue to meet ASME code requirements, the likelihood of occurrence of malfunctions evaluated in the FSAR have not been increased.

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

   ☐ Yes
   ☒ No

   BASIS:

   Regarding changes to FSAR Appendix 5.2A

   The primary and secondary safety valves are used to mitigate accidents, as described in the FSAR, when the RCPB is challenged. Release of fluid through the valves has consequence, however, the revised valve analysis only determines if the valves are still adequate to perform their design basis function, i.e., meet ASME code requirements, following EPU; the analysis itself does not change the quantity of the flow. The evaluation of any increased consequences of accidents are evaluated in the ER(s) which evaluates the changes in consequences resulting from EPU (e.g., EPU Chapter 15 Dose and AST ER-W3-2004-0276-000). Additionally, this change does not have any impact on the consequences of the accident evaluated in FSAR Section 15.6.3.4, Inadvertent Opening of a Pressurizer Safety Valve discussed in Question 1 above since the analysis does not affect the equipment. Therefore, because the change to the Overpressure Design Report does not in itself affect any accident with consequences and does no in itself change the consequence of any accident evaluated in the FSAR, this change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

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

BASIS:

Regarding changes to FSAR Appendix 5.2A

As discussed in Question 2 above, FSAR Section 15.6.3.4, Inadvertent Opening of a Pressurizer Safety Valve, evaluates an accident initiated by a malfunction of a pressurizer safety valve, however, there are no radiological consequences for this event. There are no malfunctions with radiological consequences described in the FSAR associated with the primary or secondary safety valves.

Therefore, because the change to the Overpressure Design Report does not in itself affect any malfunction with consequences evaluated in the FSAR, this change 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 FSAR.

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

BASIS:

Regarding changes to FSAR Appendix 5.2A

The revised Overpressure Protection Design Report does not make any physical changes to any SSC, does not change any operating characteristic of any SSC, does not change any setpoint for any SSC, does not change the reliability or availability of any SSC, and does not change any interfaces between important to safety SSCs. The report only verifies that the valves are adequate to perform there design basis functions, i.e., meet the ASME code requirements, following EPU. Therefore, this activity does not create the possibility for an accident of a different type than any previously evaluated in the USAR.

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

BASIS:

Regarding changes to FSAR Appendix 5.2A

The revised Overpressure Protection Design Report does not make any physical changes to any SSC, does not change any operating characteristic of any SSC, does not change any setpoint for any SSC, does not change the reliability or availability of any SSC, and does not change any interfaces between important to safety SSCs. The report only verifies that the valves are adequate to perform there design basis functions, i.e., meet the ASME code requirements, following EPU.

As long as these valves function in accordance with ASME code requirements, the validity of the safety analysis are valid. The revised overpressure protection report verifies that valves will continue to meet ASME and NRC (SRP) requirements and makes no physical changes to them. The only malfunction that was considered credible is the one described in FSAR Section 15.6.3.4, Inadvertent Opening of a Pressurizer Safety Valve. The changes to the report will not cause a different result of this malfunction. Therefore, this activity does not create the possibility of a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR since the valves will continue to function as per design, code and regulatory requirements.

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

BASIS:

Regarding changes to FSAR Appendix 5.2A

The revised Overpressure Protection Design Report does not make any physical changes to any SSC, does not change any operating characteristic of any SSC, does not change any setpoint for any SSC, does not change the reliability or availability of any SSC, and does not change any interfaces between important to safety SSCs. The report only verifies that the valves are adequate to perform there design basis functions, i.e., meet the ASME code requirements, following EPU. As long as these valves function in accordance with ASME code requirements, the design basis limits for the fission product barriers (i.e., in this case the RCPB) will not be exceeded; the report indicates that the valves are capable of fulfilling their design basis function. Additionally the analysis does not alter the design basis limit for any fission product barrier as described in the FSAR.

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

☐ Yes
☒ No

BASIS:

The information in Appendix 5.2A, "Overpressure Protection for Combustion Engineering 3410 MWt Pressurizer Reactors" is being removed from the FSAR; it is being replaced by an equivalent licensee-controlled engineering calculation (DAR-OA-03-10). Subsection NB-7300 of Section III, Division 1 of the ASME B&PV code requires that a summary technical report be prepared to document compliance. The original overpressure protection report is documented in "Nuclear Steam Supply System Overpressure Protection Report for Louisiana Power & Light Company Waterford Unit No.3", dated July 22, 1982. The intent of DAR-OA-03-10 is to provide a revised overpressure protection report for Waterford-3 to account for the changes associated with the power uprate to 3716 MWt. This report demonstrates that the existing main steam safety valves (MSSVs) and primary safety valves (PSVs) meet the requirements of the ASME B&PV Code Section III, Division 1, Nuclear Power Plant Components, 1974 Edition, up to and including summer 1975 addendum. The ASME code edition is the same design standard used in the initial overpressure protection report.

The Appendix 5.2A provided information on the engineering information that was used to establish the initial sizing of the primary and secondary safety valves. This includes core power plots, analysis for a series of loss of load events with different sized primary safety valves and defining the power level the secondary valves were capable of sustaining when passing full capacity steam flow. This initial design information is not in the EPU Overpressure Protection Report. The EPU Overpressure Protection Report presents current relevant data and demonstrates the plant continues to meet the requirements of the ASME Boiler and Pressure Vessel Code.

Code input values for the initial conditions of moderator temperature coefficient, doppler coefficient, plant pressure, pressurizer level and reactor power uncertainty continuer to be selected to produce peak primary and secondary pressures. The input values have been selected to be consistent with EPU groundrules.

The ASME code requires analysis to demonstrate safety valve blowdown values in excess of 5% will produce acceptable results. The EPU analysis evaluated additional blowdown of 10 % below the setpoint as acceptable. This value covers the range of blowdown provided in the groundrules. Reference to blowdowns greater than 10% were no longer included.

The Appendix 5.2A event was based on design code and a paragraph is included to indicate a complete loss-of-load event is described in Chapter 15. The EPU Overpressure Protection Report uses the Chapter 15 Loss of Condense Vacuum Event so a statement referencing Chapter 15 is no longer necessary.

<u>SRP Requirements</u>:

Standard Review Plan (SRP) Section 5.2.2, Overpressure Protection states that overpressure protection for the reactor coolant pressure boundary (RCPB), during power operation of the reactor, is ensured by application of relief and safety valves and the reactor protection system.

Safety valves shall be designed with sufficient capacity to limit the pressure to less than 110% of the RCPB design pressure (as specified by the ASME Boiler and Pressure Vessel Code), during the most severe abnormal operational transient with reactor scram. Also, sufficient margin shall be available to account for uncertainties in the design and operation of the plant assuming:

(1) The reactor is operating at a power level that will produce the most severe overpressurization transient.

(2) All system and core parameters are at values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.

(3) The reactor scram is initiated by the second safety-grade signal from the reactor protection system.

(4) The discharge flow is based on the rated capacities specified in the ASME Boiler and Pressure Vessel Code, for each type of valve.

<u>NRC's SER</u>

The NRC evaluated the Waterford 3 analysis for valve sizing against the requirement of SRP Section 5.2.2, during the initial licensing process and discussed the adequacy of the valve sizing, this is reported in the WF3 SER (NUREG-0787) Section 5.2.2 and SSER 3 Section 5.2.2.

SER 5.2.2 Overpressurization Protection:

"Overpressure protection of the primary coolant system is designed to accommodate both low and high temperature operation. ...The high temperature overpressure protection system is designed to maintain secondary and primary operating pressures within 110% of design by means of 2 primary safety valves, 12 secondary safety valves, and the reactor protection system. The secondary safety valves are sized to pass a steam flow equivalent to a power level of 3,580 MWt, which is greater than the proposed licensed power level of 3,410 MWt. The reactor is designed to trip at an RCS pressure of 2,400 psia while the primary pressurizer safety valves are designed to lift at a pressure of 2,500 psia, which is system design pressure." (SER Section 5.2.2)

"The design basis event for sizing the primary safety valves is loss of load with a delayed reactor trip. In the analysis provided, no credit is taken for letdown, charging, pressurizer spray, secondary bypass, or feedwater flow after turbine trip. At the onset of the transient, the RCS and main steam supply system (MSSS) are at the maximum rated output plus a 2% uncertainty. The moderator and Doppler coefficients used for the analysis maximize the pressure and power excursion." (SER Section 5.2.2)

"Under the assumptions of this analysis, a low steam generator level trip setpoint would be reached at about the same time the high pressurizer trip setpoint is reached. The peak primary and secondary system pressures are limited to 110% of design pressure during the loss-of-load transient. SRP Section 5.2.2 states that the high pressure reactor trip or second safety grade scram signal, whichever occurs later, should be used for sizing the primary system safety valves. The staff requires the applicant to confirm that this criterion is met in sizing Waterford 3 safety valves. The staff will report the resolution of this item in a supplement to this SER." (SER Section 5.2.2)

SSER 3 Section 5.2.2:

"Section 5.2.2 of the SER stated that the SRP Section 5.2.2 requires that the high pressure reactor trip or second safety-grade scram signal, whichever occurs later, should be used for sizing the primary system safety valves. The staff requires the applicant to confirm that this criterion is met in sizing Waterford 3 safety valves." (SSER 3 Section 5.2.2)

"In Amendment 25 to the FSAR, the applicant has stated that an analysis has been performed to demonstrate that the sizing of the primary system safety valves is adequate if it is assumed that the reactor is tripped on the second safety-grade trip signal. The loss of load event (the design basis event for sizing the primary safety valves) has been reanalyzed with no credit taken for Doppler feedback on core power and it was assumed that the reactor is tripped on low steam generator level, approximately eight seconds after the high pressurizer pressure trip setpoint is reached. The peak RCS pressure is well below the limit of 110 percent of design pressure. "(SSER 3 Section 5.2.2)

"Based on the above, the staff has concluded that the Waterford 3 design meets the acceptance criteria in SRP (NUREG-75/087) Section 5.2.2 and is acceptable." (SSER 3 Section 5.2.2)

Evaluation of Revised Analysis:

An evaluation of the SER section 5.2.2.1 High Temperature Overpressure Protection requirements against DAR-OA-03-10, Waterford 3 plant specific Overpressure Protection Report, and the Loss of Condenser Vacuum event (CN-TAS-02-53), the event used for verifying the adequacy of the sizing of the valves for EPU, was performed. The results of the evaluation determined the following:

- SER: 2 PSVs - CN-TAS-02-53 Paragraph 6.2.2.2 indicates that 2 valves open at 2575 psia. (SER maintained)
- SER: 12 MSSVs - CN-TAS-02-53 Paragraph 6.2.3.4 note there are 12 setpoint values. (SER maintained)
- SER: RCS high pressure trip at 2400 psia - CN-TAS-02-53 Paragraph 6.2.2.5 2422 psia is used. (SER bounded)
- SER: Pressure maintained below 110% of design Pressure - CN-TAS-02-53 PSV lift setpoint (see above) 2575 psia. This includes maximum tolerance. (SER bounded)
- SER: Design basis loss of load with a delayed reactor trip.

  The safety analysis Loss of Condenser Vacuum (LOCV) event (described in FSAR subsection 15.2.1.3) is the loss of load event that results in the highest primary and secondary pressure events.

The Feedwater Line Break is also looked at because it too can produce high primary pressures. The large feedwater line break analysis FSAR Chapter 15 results in a peak pressurizer pressure greater than the peak pressure from LOCV. However, the higher pressure is achieved due to the safety analysis methodology that unrealistically models the feedwater line at the bottom of the steam generator downcomer. The design bases accident in the EPU Overpressure Protection Report is the Loss of Condenser Vacuum.

The ASME Code (Section III, Division 1, NB-7320) specifies that the analysis consider a loss of 100% of the heat sink when the thermal output of the reactor is at 100% of its rated power. The overpressure protection analysis for EPU considers the LOCV event as the peak pressure transient. The loss of condenser vacuum causes a turbine trip (that is, a loss of load). The event also makes the steam bypass system unavailable, and results in a loss of the steam turbine driven main feedwater pumps. These additional considerations exacerbate the RCS heatup and pressurization, making the LOCV the most adverse loss of load event.

This event includes a loss of 100% heat sink, disables the reactor power cutback system, and delays the reactor trip until high pressurizer pressure trips the reactor. The LOCV scenario resulting in the peak primary system pressure the SG low level trip (which is not credited) would act before the low pressurizer pressure trip (thus meeting the SRP 5.2.2 requirement that the overpressure analysis credit the second safety-grade signal from the reactor protection system). DAR-OA-03-10 is a Waterford 3 plant specific Overpressure Protection Report that meets the ASME Boiler and Pressure Vessel Code Section III, Nuclear Vessels, 1974 Edition, up to and including summer 1975 addendum. The 10 CFR 50.55a requirement that systems and components meet the ASME B&P code has not changed.

- No credit for letdown, charging, pressurizer spray, secondary bypass or feedwater after turbine trip. (see OPPR assumption 4) (SER maintained)
- SER: Maximum rated output plus uncertainty (2%). CN-TAS-02-53, uncertainty was 2% but now is 0.5%. The increase in licensed power from 3390 MWt to 3441 MWt was supported by a reduction in the power measurement uncertainty at full power from 2% to less than 0.5% using ultrasonic flow measurement equipment for feedwater flow (via License Amendment 183). The reduced uncertainty remains valid at the power level of 3716 MWt. (SER requirement is maintained but input value has changed)
- Moderator and Doppler coefficients maximize pressure. (see OPPR assumptions 2 & 3)(SER maintained)
- Note previous OPPR low SG level trip and high pressurizer trip setpoints were set to occur at the same time. The NRC preferred the later of the two trips be used. The EPU LOCV meets SRP section 5.2.2. (see CN-TAS-02-53 section 6.1 "(if no credit is taken for a reactor trip on turbine trip, or a low steam generator level trip) a reactor trip on high pressurizer pressure occurs".
- The peak primary and secondary pressure is limited to 110% of design pressure. This is the ASME requirement. (SER maintained)
- The analyses demonstrate that the primary safety valves, secondary safety valves, and reactor protection system maintain the Waterford-3 reactor coolant system below 110% of design pressure during the limiting case transient. The adequacy of the secondary safety valves was originally demonstrated by comparing their capacity "sized to pass a steam flow equivalent to a power level of 3,580 MWt" to the 3,410 MWt rated capacity of the plant. The adequacy of the MSSVs under EPU conditions is demonstrated explicitly, by showing that the secondary pressure remains below 110% of the design pressure. This also satisfies the SRP requirement that states: "Safety valves shall be designed with sufficient capacity to limit the pressure to less than 110% of the RCPB design pressure (as specified by the ASME Boiler and Pressure Vessel Code), during the most severe abnormal operational transient with reactor scram." (SER bounded)

The loss of condenser vacuum analysis for EPU is performed using the CENTS computer code, as described in the EPU submittal. The existing overpressure protection analysis was performed using CE developed NSSS simulation codes, which were not required to be reviewed or approved by the NRC. The CENTS code includes the features of the codes used in the existing overpressure protection analysis, as described in FSAR Appendix 5.2A: the codes used include reactor kinetics, thermal and hydraulic performance of the RCS, and the thermal and hydraulic performance of the steam generators; the simulations include the effects of RCP performance, elevation heads, inertia of surge line water and friction drop in the surge line. The acceptability of using of the CENTS code is part of the EPU LAR submittal. The LAR is currently awaiting NRC approval and the approved NRC Safety Evaluation Report addressing this request must be reviewed to ensure that the assumptions made in this evaluation remain valid; see section II.A.5 of this 50.59 Review.

Conclusion:

The results of the analysis demonstrates that the safety valves have sufficient capacity to limit the pressure to less than 110% of the RCPB design pressure (as specified by the ASME Boiler and Pressure Vessel Code), during the most severe abnormal operational transient with reactor scram. Also, sufficient margin is available to account for uncertainties in the design and operation of the plant assuming:

(1) The reactor is operating at a power level that will produce the most severe overpressurization transient.

(2) All system and core parameters are at values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.

(3) The reactor scram is initiated by the second safety-grade signal from the reactor protection system.

(4) The discharge flow is based on the rated capacities specified in the ASME Boiler and Pressure Vessel Code, for each type of valve.

As demonstrated above, the results of DAR-OA-03-10 are conservative for the postulated loss of load event for EPU and utilize methodology approved by the NRC and meet the SRP Section 5.2.2 requirements. Therefore, the activity does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses and no NRC commitments are violated.

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 ENS-LI-113.