UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD
ARLINGTON TX 76011-4511

September 30, 2015

Mr. Michael R. Chisum
Site Vice President
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – NRC EVALUATIONS OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000382/2015008

Dear Mr. Chisum:

On August 27, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Waterford Steam Electric Station, Unit 3. On August 27, 2015, the NRC inspectors discussed the results of this inspection with you and other members of your staff. On September 17, 2015, the final results of the inspection were shared with your staff in a telephone exit. Inspectors documented the results of this inspection in the enclosed inspection report.

The NRC inspectors documented one finding of very low safety significance (Green) in this report. This finding involved a violation of NRC requirements. The NRC is treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2.a. of the NRC Enforcement Policy.

Further, inspectors documented two licensee-identified violations which were determined to be of very low safety significance. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of the non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Waterford Steam Electric Station, Unit 3.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, “Public Inspections, Exemptions, Requests for Withholding,” a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC

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Sincerely,

/RA/

Thomas R. Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-382
License No. NPF-38

Enclosure:
Inspection Report 05000382/2015-008
w/ Attachment: Supplemental Information

cc w/ encl: Electronic Distribution

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Letter to Michael R. Chisum from Thomas R. Farnholtz, dated September 30, 2015

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – NRC EVALUATIONS OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000382/2015008

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000382
License:	NPF-38
Report:	05000382/2015008
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	17265 River Road Killona, LA 70057
Dates:	August 10 through August 27, 2015
Inspectors:	M. Williams, Reactor Inspector W. Sifre, Senior Reactor Inspector R. Kopriva, Senior Reactor Inspector
Approved By:	Thomas R. Farnholtz Branch Chief Division of Reactor Safety

SUMMARY

IR 05000382/2015008; 08/10/2015 – 08/27/2015; Waterford Steam Electric Station, Unit 3; Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications.

This report covers a two-week announced baseline inspection on evaluations of changes, tests, and experiments and permanent plant modifications. The inspection was conducted by Region IV based engineering inspectors. One finding, with two examples, was identified. The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0310, "Aspects Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5, dated February 2014.

A. NRC-Identified Findings and Self-Revealed Findings

Cornerstone: Mitigating Systems

Green. The team identified two examples of a Green, non-cited violation of Technical Specification 6.8.1, which states, in part, "Written procedures shall be established, implemented, and maintained, covering the activities including procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A.6.w, Acts of Nature (e.g., tornado, flood, damn failure, earthquakes)." Specifically, in the first example, prior to August 27, 2015, the licensee failed to establish adequate procedures to ensure the manual actions required within specified time limits can be completed before full draindown of the ultimate heat sink (wet cooling tower basins) after a tornado event. In the second example, prior to August 27, 2015, the licensee failed to establish adequate procedures to clarify whether the main steam isolation valve area was considered outdoors and therefore subject to the requirements for unmonitored items stored in the protected area. Unsecured scaffold material stored in this area had not been evaluated for potential to become projectiles and endangering nearby safety-related equipment during high winds. In response to this issue, the licensee inspected the area and secured all loose debris. This finding was entered into the licensee's corrective action program as Condition Reports CR-WF3-2015-05624 and CR-WF3-2015-05601.

The team determined that the failure to maintain adequate procedures to ensure compliance with technical specifications and Regulatory Guide 1.33 was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to establish adequate procedures to ensure availability of mitigating equipment during and after an event involving acts of nature. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems

Screening Questions." The issue screened to Exhibit 4, "External Events Screening Questions," because both examples involved a design basis tornado. Per Exhibit 4, the issue screened to a more detailed risk evaluation because: 1) the first issue could starve safety systems of water, failing the safety function, and 2) the second issue could cause a plant trip and a loss of condenser heat sink initiating event. Therefore, the Region IV senior reactor analyst performed a more detailed risk evaluation that included both issues. Given that there was no change in core damage frequency for the first issue, and the change in core damage frequency for the second example was 1.2×10^{-9} per year, combined, the analyst determined that the finding was of very low safety significance (Green). This finding had a cross-cutting aspect in the area of problem identification and resolution, evaluation, because the licensee failed to thoroughly evaluate issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance (P.2).

B. **Licensee-Identified Violations**

Two violations of very low safety significance that were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and associated corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R17 Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications (71111.17T)

.1 Evaluations of Changes, Tests, and Experiments

a. Inspection Scope

The inspectors reviewed seven evaluations performed pursuant to Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 59, to determine whether the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 19 screenings, where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, and experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issue requiring the change, tests and experiment was resolved
- the licensee conclusions for evaluations of changes, tests, and experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The list of evaluations, screenings and/or applicability determinations reviewed by the inspectors is included as an Attachment to this report.

This inspection constituted 7 samples of evaluations and 19 samples of screenings and/or applicability determinations as defined in IP 71111.17-04.

b. Findings

Introduction. The team identified two examples of a Green, non-cited violation of Technical Specification 6.8.1. Specifically, in the first example, the licensee failed to establish adequate procedures to ensure the manual actions required within specified time limits can be completed before full draindown of the ultimate heat sink (wet cooling tower basins) after a tornado event. In the second example, the licensee failed to establish adequate procedures to clarify whether the main steam isolation valve area was considered outdoors and therefore subject to the requirements for unmonitored items stored in the protected area. Unsecured scaffold material stored in this area had not been evaluated for potential to become projectiles and endangering nearby safety-related equipment during high winds.

Description. The team reviewed Procedure EP-002-100, “Technical Support Center Activations, Operation and Deactivation,” for station requirements during a design basis tornado event. Several steps require manual actions within specified timeframes in order to prevent full draindown of the wet cooling towers, which is the station’s ultimate heat sink. The procedure lacked details for access to structures, storage of equipment and tools required, specific training necessary for operators, and coordination with the control room and security. Without such details specified in the procedure, there was not a complete and accurate evaluation to validate the manual actions required could be completed within specified times.

During a walkdown of the main steam isolation valve area, the inspectors observed a large scaffold with unsecured scaffold building material stored on several of its platforms. The main steam isolation valve area is protected on its sides with vertical walls, but is open overhead to the environment and protected from external missile projectiles with large grating. The team reviewed Procedure UNT-007-060, “Control of Loose Items,” which includes requirements for unmonitored items stored outdoors in the protected area, and specifically scaffold material. The licensee was unsure if the main steam isolation valve area was considered outdoors, and did not have an evaluation to justify that storage of loose debris in the main steam isolation valve rooms would not be subject to the influence of wind loading and become potential missiles projectiles endangering nearby safety-related equipment.

Analysis. The team determined that the failure to maintain adequate procedures to ensure compliance with technical specifications and Regulatory Guide 1.33 was a performance deficiency. This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to establish adequate procedures to ensure availability of mitigating equipment during and after an event involving acts of nature. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, Exhibit 2, “Mitigating Systems Screening Questions.” The issue screened to Exhibit 4, “External Events Screening Questions,” because both examples involved a design basis tornado. Per Exhibit 4, the issue screened to a more detailed risk evaluation because: 1) the first issue could starve safety systems of water, failing the safety function; and 2) the second issue could cause a plant trip and a loss of condenser heat sink initiating event. Therefore, the Region IV senior reactor analyst performed a more detailed risk evaluation that included both issues. Given that there was no change in core damage frequency for the first issue, and the change in core damage frequency for the second example was 1.2 x 10^{-9} per year, combined, the analyst determined that the finding was of very low safety significance (Green). This finding had a cross-cutting aspect in the area of problem identification and resolution, evaluation, because the licensee failed to thoroughly evaluate issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance (P.2).

The finding involved two sub-issues including: 1) the failure to provide adequate instructions for an operator action that was required 45 hours after the initiating event; and 2) an inadequate procedure that resulted in an unanalyzed condition of loose debris in the main steam isolation valve area. Both issues involved a design basis tornado.

Operator Action Issue: In all, the affected procedure covered four operator actions. Each included the alignment of different water supplies to the plant. However, the instructions covering the first three actions were adequate. The fourth operator action that included the use of a temporary sump pump to bring water into the plant would not be required during the first 44 hours of the initiating event. While design basis accidents can last as long as 30 days, the probabilistic risk assessment (PRA) mission time is only 24 hours. This is based on the expectation that the licensee will have abundant resources available to mitigate an accident within this time period. The licensee is required to meet the design basis requirements. However, the 24-hour mission time is only considered as part of the risk assessment. Since the fourth operator action would be required more than 24 hours after the start of the initiating event, this issue was of very low safety significance (Green).

Debris in Main Steam Isolation Valve Area Issue: The analyst determined the change to the core damage frequency (ΔCDF) associated with loose debris in the main steam isolation valve area. During a tornado, the debris could cause damage to the main steam isolation valves or subcomponents. The worst case event would include a plant trip and main steam isolation valve closure. This would result in isolating the main condenser as a viable heat sink. Alternatively, if a main steam isolation valve were damaged such that it stuck open, the main condenser would remain viable, which would be the preferred condition. Therefore, stuck open main steam isolation valves during a tornado were not a risk concern. The analyst made the following assumptions:

- The average number of tornados in Louisiana per year was 27 (see http://www.erh.noaa.gov/cae/svrwx/tornadobystate.htm).

- The total area for the state of Louisiana was 51,840 square miles (see http://www.enchantedlearning.com/usa/states/area.shtml). The frequency for a tornado per square mile of land was 27/51,840 = 5.2E-4/year.

- The analyst conservatively based the evaluation on a one square mile area that included the main steam isolation valve area. If a tornado hit this area, the analyst assumed that the main steam isolation valves would fail closed. This would cause a loss of condenser heat sink and a plant trip.

- The analyst assumed that the condenser was not recoverable.

- The analyst assumed that the tornado would not cause a loss of offsite power. A tornado that caused a loss of offsite power would cause a plant trip and loss of condenser heat sink anyway, so the debris in the main steam isolation valve area would have no effect on the plant response.

Calculations: The analyst used the NRC's Waterford-3 Standardized Plant Analysis Risk (SPAR) model, Revision 8.16, with a truncation limit of E-11, to evaluate this finding.

The base case conditional core damage probability was 1.4E-7. This was the case without the performance deficiency. The current case conditional core damage probability was 2.4E-6. This was the case with the performance deficiency. Therefore the incremental conditional core damage probability was 2.3E-6.

The change to the core damage frequency was the tornado frequency multiplied by the incremental conditional core damage probability.

ΔCDF = 5.2E-4/year * 2.3E-6 = 1.2E-9/year

The dominant core damage sequences included the tornado induced closure of the main steam isolation valves, which would cause a plant trip and a loss of condenser heat sink. The low tornado frequency and the unaffected power operated relief valves helped to minimize the risk.

Total ΔCDF: The total risk for both issues is the sum of the ΔCDF values. However, the first issue did not have a quantifiable ΔCDF. Therefore, the ΔCDF for both issues combined was approximately 1.2E-9/year. The finding was of very low safety significance (Green).

Since the ΔCDF was less than 1E-7, no evaluation of the large early release frequency was required.

Therefore, the issue was determined to have very low safety significance (Green).

Enforcement. The team identified two examples of a Green, non-cited violation of Technical Specification 6.8.1, which states, in part, "Written procedures shall be established, implemented, and maintained, covering the activities including procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A.6.w, Acts of Nature (e.g., tornado, flood, damn failure, earthquakes)." Contrary to the above, prior to August 27, 2015, the licensee failed to establish adequate procedures covering activities related to acts of nature. Specifically, in the first example, the licensee failed to establish adequate procedures to ensure the manual actions required within specified time limits can be completed before full draindown of the ultimate heat sink (wet cooling tower basins) after a tornado event. In the second example, the licensee failed to establish adequate procedures to clarify whether the main steam isolation valve area was considered outdoors and therefore subject to the requirements for unmonitored items stored in the protected area. Unsecured scaffold material stored in this area had not been evaluated for potential to become projectiles and endangering nearby safety-related equipment during high winds. In response to this issue, the licensee inspected the area and secured all loose debris. This finding was entered into the licensee's corrective action program as Condition Reports CR-WF3-2015-05624 and CR-WF3-2015-05601. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 5000382/2015008-001, "Inadequate Procedures for Design Basis Tornado Event."

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed 11 permanent plant modifications that had been installed in the plant during the last three years. The modifications were selected based upon risk significance, safety significance, and complexity. The inspectors reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;
- the changes were in accordance with the specified design requirements;
- the procedures and training plans affected by the modification have been adequately updated;
- the test documentation as required by the applicable test programs has been updated; and
- post-modification testing adequately verified system operability and/or functionality.
- The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an Attachment to this report.

This inspection constituted 11 permanent plant modification samples as defined in IP 71111.17 04.

.2.1 Allow ¾ inch Schedule 80 Pipe as Equivalent to ¾ inch Schedule 40 Pipe Used on Actuator for FW-173A(B) to Help Prevent Future Failures Similar to that Experienced Coming Out of RF 18

The inspectors reviewed Engineering Change Package EC-42165, implemented to replace the original main feedwater regulation valve "A" actuator ¾ inch Schedule 40 carbon steel pipe nipple with a 316 stainless steel schedule 80 or 160 pipe nipple. The change provides an alternative material with improved strength and fatigue properties.

When the licensee was finishing their Refuel Outage 18 and preparing to perform a plant start-up, a ¾ inch, schedule 40, airline pipe failed on feedwater valve FW-173A. This resulted in a plant trip that was documented in Condition Report CR-WF3-2013-00445. A second failure was observed at this location and was documented in Condition Report CR-WF3-2013-00523. The feedwater regulating valves are augmented quality related and seismic category 1. They are Quality Group D valves constructed to B31.1 standards. The associated air piping is of the same quality and construction as the valve. The valves provide a safety-related backup isolation function to the main feedwater isolation valves and are required to close in five seconds or less. The location that the airline failed was in the threads of the pipe nipple that connects to the cylinder and is connected on the other end to the pneumatic valve in line to the quick exhaust valve. This engineering change package evaluated the acceptability of using schedule 80 or 160 stainless steel pipe in lieu of the installed schedule 40 pipe. Additionally, the change would shorten the piping length between the solenoid and quick exhaust. The changes provided a more robust design to mitigate the potential for fatigue failure.

The package also found it acceptable to replace the ¾ inch schedule 40 pipe on the actuator for FW-173A(B) with schedule 80 or 160 pipe. The schedule 80 or 160 pipe would be more robust to resist failure and would allow for more cross sectional material to remain when the threads are cut. It was also found acceptable to reduce the length of the nipples in the line such that the total length of the line is reduced by approximately 1 to 2 inches. Finally, it was acceptable to use a stainless steel SA312-TP316 material in lieu of the A36 material originally specified. The impacts on air flow and weight are small such that they will have no adverse impact on valve performance or seismic qualification. The inspectors did not identify any concerns with the design change package.

.2.2 Replace Emergency Diesel Generator Air (EGA) Start Air Compressors due to Reliability Concerns

The inspectors reviewed Engineering Change Package EC-43474, implemented to replace the emergency diesel generator air start air compressor assembly (air compressor and electric motor). The air start compressor assemblies have shown an adverse trend in performance and reliability over the past years as documented in condition report documents, and cited specifically in Condition Report CR-WF3-2013-01185. An evaluation for common cause of air compressor and cooler gaskets repairs performed in Condition Report CR-WF3-2011-06560 indicated a history that the Kellogg A352TV air compressors did not perform reliably. The Kellogg A352TV compressor line has been discontinued and the replacement alternative air compressor (manufactured to Kellogg specifications) is also not performing with adequate reliability. Peer check within the nuclear industry identified use of Quincy air compressor model QR-25 at the McGuire and Cooper nuclear stations have demonstrated the ability to provide good reliability at both stations under similar service in emergency diesel generator air start systems.

Engineering Change Package EC-43474 was initiated to resolve the reliability problem, and the unavailability of parts and service, of the original Kellogg air compressor model A352TV, A352TVX, or R352HU. The installation of a Quincy air compressor model QR-25 D350HP would provide greater air volume than the design specified compressed air volume of 32.2 standard cubic feet per minute at a pressure of 250 pounds per square inch. The Quincy air compressor is currently manufactured with technical support and replacement parts readily available. Regarding reliability, the Quincy model air compressor is currently installed at two other nuclear stations, and has performed well, with no concerns pertaining to reliability. The power requirements for the Quincy model air compressor 15 horse power electric motor are met by the existing electrical cabling and fuses that supply the existing 10 horse power air compressor electric motor power. The control system use of pressure switches to control operation of the Quincy model air compressor is compatible with the existing magnetic motor starter system currently installed. The inlet air temperature limit to the air dryer is 120 degrees Fahrenheit. This would be met by the inter-cooler that cools the compressed air between the first and second stages of the Quincy model air compressor. The implementation of this engineering change would be performed by this parent Engineering Change Package EC-43474 and four separate engineering change packages to control implementation of component change, one for each air start air compressor: Engineering Change Packages EC-45362 for EGAMCMP0001A, EC-45363 for EGAMCMP0002A, EC-45364 for EGAMCMP0001B, and EC-45365 for EGAMCMP0002B. The inspectors did not identify any concerns with the design change package.

.2.3 Emergency Generator Fuel Standby Fuel Oil Booster Pump Head Gasket O-Ring Equivalency

The inspectors reviewed Engineering Change Package EC-48357, implemented to allow the substitution of a Viton O-Ring for the Buna N O-Ring currently installed in the Emergency Diesel Generator B Motor Driven Fuel Oil Standby Pump B (EGRMPMP0002 B). Condition Reports CR-WF3-2013-05962 and CR-WF3-2013-06020 identified a fuel leak on the Motor Driven Fuel Oil Standby Pump B (EGRMPMP0002 B). The primary leakage point was determined to be from the head gasket O-ring. The O-ring was documented as original equipment manufacturers part number 2-511-001-860 and was made of Buna N material. The O-ring was not in stock in the Waterford warehouse system. After searching the RAPID Database (inventory system for all Nuclear Utilities) spare O-rings were located, and were shipped to Waterford. At the time, the site also pursued an alternate success path, and a replacement VITON O-ring was located on the site that matched the physical dimensions of the O-ring (Parker 2-142). The engineering change supports the substitution of the VITON O-ring for the Buna N O-ring.

The replacement of the head gasket O-ring would not impact the pump's performance. The difference is considered an upgrade from Buna N material to VITON. Reviewing the critical characteristics of the O-ring, based on physical measurements of the replacement O-ring, the Parker 2-142 was a direct replacement. The site had safety-related O-rings in stock listed as Parker 2-142; however, they were made with VITON instead of Buna N material. VITON is chemical and fuel resistant to all fuels, blends of fuels, and fuel treatment additives. In fact, if operating with a low sulfur diesel fuel oil, operating experience has demonstrated that VITON is superior to Buna N. A search of industry related O-ring failures involved with fuel oil determined that Buna N material can fail in a low sulfur fuel oil environment. The inspectors did not identify any concerns with the design change package.

.2.4 Feedwater Snubber Replacement Inside Containment from PSA to LISEGA

The inspectors reviewed Engineering Change Package EC-55547, implemented to replace existing PSA Mechanical Snubbers FWSR-2A, 2B; FWSR-6A, 6B; FWSR-8A, 8B; FWSR-11; FWSR-14A, 14B; FWSR-18A, 18B; FWSR-20A, 20B; and FWSR-23 with equivalent Lisega hydraulic snubbers to improve the long term reliability of the snubbers experiencing feedwater system vibration. Condition Reports CR-WF3-2013-00565, CR-WF3-2013-00445, and CR-WF3-2014-02006 identified an increase in vibration between the main feedwater regulating valve and the associated steam generators after the replacement steam generator project was implemented. The condition reports address plant trips and failure of various snubbers and regulating valve component failures in the feedwater system as a result of the increased vibration. Engineering Change EC-44028 replaced several PSA mechanical snubbers with Lisega hydraulic snubbers during Refueling Outage 19. A Lisega snubber was added on the emergency feedwater line in the west wing area to address the increased vibration issues following the replacement of the steam generators at Waterford. It was noted that Lisega hydraulic snubbers help absorb feedwater system vibration to a certain extent and showed more resistance to the high vibration and did not lock out. Based on the past performance and industry experience, this engineering change was created to replace existing PSA mechanical snubbers FWSR-2A, 2B; FWSR-6A, 6B; FWSR-8A, 8B; FWSR-11; FWSR-14A, 14B; FWSR-18A, 18B; FWSR-20A, 20B; and FWSR-23 with equivalent Lisega hydraulic snubbers to improve the long term reliability of the snubbers experiencing feedwater system vibration.

To accommodate the new hydraulic snubbers, only PSA mechanical snubber attachments will be replaced with Lisega snubber components. The pipe clamps will be kept. All associated drawings and other design documents were revised based on the change in snubber design. The eight supports are safety-related, Seismic Class I, and support piping in the feedwater system which is located inside the reactor containment building. Per Engineering Standard PS-S-002-W, safety-related, Seismic Class I sections of the feedwater system require Seismic Class I supports. The replacement snubbers are safety-related, Seismic Class I supports, and meet the pertinent quality assurance requirements of the Quality Assurance Program. Therefore, this engineering change was listed as quality-related. The new hydraulic snubbers are qualified for the required support rating for the associated piping. Each of the hanger calculations, FWSR-2, FWSR-6, FWSR-8, FWSR-11, FWSR-14, FWSR-18, FWSR- 20, and FWSR-23, provide the normal and maximum design (faulted) load conditions for each snubber, which is taken from Pipe Stress Calculations SA1031 and SA1032. The new replacement hydraulic snubber ratings exceed the requirements for the normal and maximum design (faulted) load conditions. Hanger Calculations FWSR-2, FWSR- 6, FWSR-8, FWSR-11, FWSR-14, FWSR-18, FWSR-20, and FWSR-23 were evaluated and found that the installation of the new hydraulic snubbers will not impact the technical qualification in the hanger calculations. Engineering change markups qualifying the new hydraulic snubbers and justifying why the technical qualification of each calculation is not impacted by this equivalent replacement are provided in their respective child engineering changes. The inspectors did not identify any concerns with the design change package.

.2.5 Modify Essential Feedwater 223A Valve Booster Relay Located at +46 Elevation in the Reactor Auxiliary Building West Wing Area to Address Abnormal Venting Due to Vibration

The inspectors reviewed Engineering Change Package EC-49854, implemented to address vibration issues causing the booster relay on essential feedwater valve 223A to vibrate and leak. Operators identified air weeping from the booster relay and determined that the issue may have contributed to an equipment failure identified in Condition Reports CR-WF3-2014-00385. The modification entailed the moving of the model 61H volume booster relay from its original location on the valve actuator to a new support approximately one foot north of the actuator. This modification also required the replacement of the stainless steel tubing with stainless steel flex hose. The installation of the flex hose required a re-evaluation of the air accumulator capacity to ensure that the valve actuator design was not challenged. This modification achieved the desired outcome of decoupling the vibration sensitive relay from the vibration of the essential feedwater system.

The inspectors reviewed the testing performed to validate that the valve actuator had sufficient air to meet its design basis. Specifically the licensee verified that the change from tubing to flex hose and associated change in air volume usage did not adversely affect the air pressure and volume supplied to the valve actuator or the associated setpoints. The inspectors did not identify any concerns with the design change package.

.2.6 Weld Repair of the Seal Ring on Safety Injection Valve SI-401A

The inspectors reviewed Engineering Change Package EC-50591, implemented to repair the seal ring on safety injection valve SI-401A. The design change package involved the repair of the upper portion of the seal ring which was inadvertently ground off during a maintenance activity. This repair involved the installation of the upper portion of a spare seal ring via a partial penetration weld to the remaining portion of the existing seal ring. The intent of this modification was to repair the seal ring without affecting the original valve body design. The modification included an evaluation of the repair as equivalent to the original seal ring in form, fit, and function. The seal ring is considered an ASME Section XI pressure retaining component. The licensee performed the repair in accordance with ASME Section XI Repair/Replacement IWA-4120A.

The inspectors reviewed the modification package and associated testing and validation performed to ensure a leak tight seal. The inspectors did not identify any concerns with the design change package.

.2.7 Evaluate the New Redesigned Veritek Low Level (VLL) Boards for Replacing the Obsolete Boards Presently Being Used in High Critical Applications

The inspectors reviewed Engineering Change Package EC-53725, implemented to replace obsolete boards in the process analog control system. The process analog control system measures various plant process parameters in the field and converts those measurements into electrical signals. These signals are then routed to various devices and components for indication, control, and protection purposes. The system uses thermocouples to sense critical temperature parameters in the plant. These signals are then sent to amplifier cards and bistable modules to monitor temperatures and produce indications, annunciations, and actuation signals. The intent of this modification was to replace cards which were considered obsolete and no longer available from the vendor.

The inspectors reviewed the modification package and associated testing and validation performed to ensure that the new cards were the same as the original cards in form fit and function. The licensee also verified that the cards did not introduce any new failure modes. The inspectors did not identify any concerns with the design change package.

.2.8 Vital and Measurement Static Uninterruptible Power Supply Upgrade Project

The inspectors reviewed Engineering Change Package EC-43927, implemented to provide reliable, uninterruptible 120 VAC power to plant protection system, engineered safety features, and other safety-related loads. The modification includes the installation of two "swing" static uninterruptable power supplies that are used to transfer static uninterruptable power supply output power for the power distribution panels from the inservice static uninterruptable power supplies to swing static uninterruptable power supply. The work was completed in phases, during both online operations and outages, as an enhancement for additional isolation capability during testing and maintenance. The modification included installation of transfer switch electrical control panels to facilitate the electrical transfer or "swing" capability. The electrical control panels receive power from either the normal static uninterruptable power supply or the swing static uninterruptable power supply, and transfer that power to the associated power distribution panels. A total of six electrical control panels are installed to support three channels on each of the two divisions.

The inspectors reviewed the design package, discussed the change with the project engineer, and walked down the new equipment. The inspectors did not identify any concerns with the design change package.

.2.9 Repair Intake Structure/Weir Pile

The inspectors reviewed Engineering Change Package EC-48179, implemented to install a clamp on a cut pile to materially rejoin the piling section across the cut at the mud line. During implementation of Condition Report CR-WF3-2013-02566, corrective actions to remove damaged dolphins from the river, the wrong knee brace battered pile was accidentally cut by 80 percent, due to human error as a result of extremely muddy river water with minimal visibility for divers. The design package included calculations to support that the clamp was adequate for transferring the load from one side of the cut pile to the other. The dolphins and battered piles do not provide a safety-related function to the intake structure, but are installed in the river waterway to protect the intake structure from watercraft navigating the river.

The inspectors reviewed the design package, discussed the change with the project engineer, and visited the intake structure to visually observe the location of the repaired pile. The inspectors did not identify any concerns with the design change package.

.2.10 Evaluate Impact of Corroded Anchor Bolts on Intake Structure Rated Crane Capacity

The inspectors reviewed Engineering Change Packages EC-53733, implemented to evaluate existing capacity of the overhead travelling crane at the intake structure. The 40-ton crane is used for moving heavy equipment at the intake structure. Due to the extreme environment at the river's edge, several of the anchor bolts were found to have varying degrees of corrosion. An inspection in 2004 documented the as-found conditions of the anchor bolts at both vertical and inclined columns. Evaluation ER-W3-2004-0217 concluded that an administrative limit of 20 tons would be placed on use of the crane until repair or replacement of the corroded anchor bolts was completed. The licensee's recommendation was that the repair and replacement be completed within six months.

The inspectors reviewed the modification package, discussed it with the project engineer, and walked down the degraded anchor bolts. The inspectors did not identify any concerns with the design change package.

.2.11 Broken Anchor Bolt on Wall Plate for CCRR-1183

The inspectors reviewed Engineering Change Package EC-48021, implemented to replace a failed anchor bolt base plate in Dry Cooling Tower B. While preparing to paint piping in the area, an anchor bolt in the piping support for a component cooling water pipe sheared off flush with the base plate. The existing anchors were abandoned in place, and a design change was completed to install a new base plate for the safety-related hanger supporting the design load of the component cooling water system piping. The dry cooling towers are the atmospheric release for heat from the component cooling water system, and therefore a very moist environment susceptible to corrosion.

The inspectors reviewed the design package, discussed the change with the project engineer, and visually inspected the new anchor bolts and base plate. The inspectors did not identify any concerns with the design change package.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

.1 Review of Corrective Action Program Documents

The inspectors reviewed corrective action program documents that identified or were related to 10 CFR 50.59 program and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations of changes, tests, and experiments. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The list of specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

4OA6 Meetings

Exit Meeting Summary

On August 27, 2015, the inspectors presented the initial inspection results to Mr. M. Chisum, Site Vice President, and other members of the licensee’s staff. The licensee acknowledged the results as presented. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

On September 17, 2015, the inspectors presented the final inspection results to Mr. B. Lanka, Engineering Director, and other members of the licensee’s staff via telephone exit. The licensee acknowledged the results as presented. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

4OA7 Licensee-Identified Violation(s)

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy for being dispositioned as non-cited violations.

- The licensee failed to establish the alarm setpoints for the pressurizer relief valve tailpipe temperature in accordance with their design basis. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion III, Design Control, states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, prior to November 7, 2011, the licensee failed to assure that the design basis for the pressurizer relief valve tailpipe alarm setpoints were correctly translated to specifications and procedures. Specifically, the pressurizer relief valve tailpipe setpoint was set at 280 degrees Fahrenheit in accordance with the setpoint study documented prior to initial startup. The licensee failed to adjust the setpoint to the design basis setting of the actual system temperature plus 25 degrees Fahrenheit as specified. This finding was assessed using Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings at-Power," and was determined to be of very low safety significance (Green) because it did not result in the loss of a system or function. This issue was entered into the licensee’s corrective action program as Condition Report 2015-05607.

- The licensee failed to perform a full 10 CFR 50.59 evaluation for changes made to the Core Operating Limits Report during Refueling Cycle 20. Title 10 of the Code of Federal Regulations, Part 50.59, states, in part, that licensees may make changes to the facility as described in the final safety analysis report without obtaining a license amendment only if the change does not meet any of the criteria in paragraph (c)(2), which includes a full evaluation of eight criteria supporting the conclusion that the change is not adverse. Several physics assessment checklist exceptions for Cycle 20 inputs were more limiting as compared to those in the licensing basis analysis of record, including the axial shape index limit and azimuthal tilt. Contrary to the above, prior to August 27, 2015, the licensee failed to complete a full 10 CFR 50.59 evaluation for changes to the core operating limits report for Cycle 20. Specifically, the axial shape index limit is an input to various Reload Physics and Accident analyses, and is specifically discussed in the updated final safety analysis report, while the azimuthal tilt had two inputs related to heating of reactor vessel internals that should have been considered adverse and therefore required a full 10 CFR 50.59 evaluation. This finding was assessed using Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings at Power,” and was determined to be of very low safety significance (Green) because it did not result in the mismanagement of reactivity by operators. This issue was entered into the licensee’s corrective action program as Condition Reports CR-WF3-2015-04040, CR-WF3-2015-04045, and CR-HQN-2015-00684.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Chisum, Site Vice President, Operations
M. Barreto, Senior Engineer
D. Becker, Technical Specialist
C. Bergeron, Senior Engineer
L. Brown, Specialist, Regulatory Assurance
K. Dolese, Senior Engineer
T. Fliescher, Senior Lead Engineer
R. Gabb, Engineer I
D. Galodoro, Senior Lead Engineer
A. Griffin, Senior Engineer
M. Groome, Senior Lead Engineer
T. Hemple, Technical Specialist
M. Haydel, Manager, Design & Program Engineering
C. Lunk, Senior Engineer
S. Meiklejohn, Licensing Specialist Senior
L. Milster, Senior Licensing Engineer
S. Munchi, Senior Lead Engineer
M. Peno, Senior Engineer
D. Phillips, Technical Specialist IV
S. Picard, Engineer II
C. Pickering, Senior Engineer
M. Progler, Senior Engineer
G. Settoon, Engineer I
B. Steelman, Senior Engineer
A. Tojeiro, Engineer I
R. Tran, Senior Engineer

NRC Personnel

F. Ramirez, Senior Resident Inspector
C. Speer, Resident Inspector
M. Orenak, Project Manager, NRR
A. Wang, Project Manager, NRR

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000382-2015-008-01 NCV (Section 1R17)

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

EC - 08434	EC- 25804	EC- 41099
EC –19258	EC- 29304	EC - 55148
EC –35812	EC- 31375	EC - 57152
EC –41749	EC- 31826	EC - 02298
EC –42745	EC- 35111	EC - 53675
EC –47327	EC- 40826	EC - 08465
EC –48521	EC- 25804	EC - 051582
EC - 48179	EC - 45550	EC – 56135
EC – 53733	EC - 48021	EC - 49378
EC – 55022	EC - 47301	

10 CFR 50.59 Evaluations

Number	Description or Title	Revision
2013-01	EC-43821, Provide Regulated N2 Supply to SIT 2B	0
2015-01	CR-WF3-2015-3565 Operability/Compensatory Measure [Change Emergency Feedwater backup flow control valves (EFW-223A9B)] from automatic to manual operation.	1
2015-02	Fukushima EP Communications – EC 47846 and LBDCR 15-019	0
2010-06	EC 14765, SI-405A(B) Bypass Fill / Equalization Line Addition / ECN 25944, Changes for Calculation MPR-2390 R3 SDC Gas Intrusion Analysis	1
2011-02	ECN 25944, Changes to EC14765 for Calculation MPR-2390 Revision 3 SDC Gas Intrusion Analysis	1
2014-01	EC – 43927 Vital and Measurement SUPS Upgrade Project	0
2014-02	EP-002-100 Technical Support Center Activation, Operation, and Deactivation	42

Miscellaneous

Number	Description or Title	Revision/ Date
LO-WLO-2015-00038	50.59 and Permanent Plant Modifications – Focused Assessment (Pre-NRC Inspection 6/15/15-6/18/15)	0
NEI 96-07, Appendix B	Guidelines For 10 CFR 72.48 Implementation	March 5, 2001
WLP-OPS-122SI	2012 Cycle 2 SI with SOER 97-01	1
W3F1-2010-0019	Technical Specification Table 3.4-1 Isolation Valve Addition Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38	February 22, 2010
CDCC-21900	Instrument Setpoint Study	September 4, 1981
ANSI/ANS-58.9-1981	Single Failure Criteria for Light Water Reactors.	February 17, 1981
ASME OM-3	Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems.	1982
CN-TAS-08-40 R0	Waterford 3 Feedwater Line Break Analysis for Replacement Steam Generators.	June 4, 2009
EC-17442	Evaluate the Use of the RCP Motor Upender Top Plate (5817-10935) For Lifting of the RCP Motor.	0
EC-43821	Provide Regulated Nitrogen Supply to SIT 2B.	0
EC-47846	Fukushima E-Plan Communications.	July 2, 2015
EC-55548	Child EC 1 – Replace FWSR 11 PSA Snubber with Comparable LISEGA Snubber on FW System Inside Containment.	0
EC-55553	Child EC 6 – Replace FWSR 2 A & B PSA Snubber with Comparable LISEGA Snubber on FW System Inside Containment	0
EC-58184	Clarify the Basis of the Op-Eval for CR-WF3-2015-3565 to Address Concerns Identified on CR-WF3-2015-3827	June 11, 2015
FWSR-2	Feedwater Safety-Related Hanger Calculation	3
LO-WLO-2015-00038	50.59 and Permanent Plant Modifications – Focused Assessment (Pre-NRC Inspection 6/15/15-6/18/15)	May 28, 2015
LO-WLO-2015-00048, CA-4	Include an Objective in the Semi-Annual Assessment to Evaluate the Effectiveness of the PAD Quality Checklist.	August 4, 2015

Miscellaneous

Number	Description or Title	Revision/ Date
LTR-LAM-13-31, Rev. 1	Evaluation of a Continuous Nitrogen Supply to Waterford 3 SIT 2B during LOCA	April 18, 2013
LTR-LIS-08-543	PWROG Position Paper on Non-condensable Gas Voids in ECCS Piping: Qualitative Engineering Judgement of Potential Effects on Reactor Coolant System Transients Including chapter 15 Events, Task 3 of PA-SEE-450.	August 19, 2008
NEI 12-01	Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities	0
NRC Information Notice 97-78	Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times.	October 23, 1997
NRC Regulatory Issue Summary 2005-20	Revision to NRC Inspection Manual Part 9900 Technical Guidance, Operability Determinations and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality of Safety.	Revision 1 April 16, 2008
Regulatory Guide 1.53	Application Of The Single-Failure Criterion To Nuclear Power Plant Protection Systems	0
SA-1031	Stress Analysis of Feedwater Piping from Pen. 3 to Steam Generator # 1.	3
SQ-MN-087	Seismic Qualification Review Team (SQRT) File # SQ-MN-087, Valve Anchor Darling 14 inch.	2

Corrective Action Program Documents (Issued)

CR-WF3-2011-01529	CR-WF3-2011-01714	CR-WF3-2013-00445	CR-WF3-2015-04806
CR-WF3-2010-03645	CR-WF3-2011-01880	CR-WF3-2013-00565	CR-WF3-2015-05543
CR-WF3-2008-04161	CR-WF3-2011-01882	CR-WF3-2013-02016	CR-WF3-2015-05252
CR-WF3-2010-03933	CR-WF3-2011-02246	CR-WF3-2013-05962	CR-WF3-2015-05580
CR-WF3-2015-05607	CR-WF3-2011-06560	CR-WF3-2013-06020	CR-WF3-2015-05624
CR-WF3-2013-05582	CR-WF3-2011-07395	CR-WF3-2014-02006	CR-WF3-2015-05601
CR-WF3-2013-05595	CR-WF3-2011-17714	CR-WF3-2015-00874	CR-WF3-2015-00684
CR-WF3-2014-00385	CR-WF3-2011-01714	CR-WF3-2015-00968	

Calculations

Number	Description or Title	Revision

Calculations

Number	Description or Title	Revision
ECMO3-003	Shutdown Cooling Operation with Suction Piping Air Intrusion	0
SQ-MN-35	Valve Masoneilon 4"	4
FWRR-6022	EFW 223A Booster Relay support	0
ECM88-024	Nitrogen Accumulator V, VIII, IX, and X Sizing Calculation.	0
ECM98-006	SIS Relief Valve Capacity and Setpoint Verification	0
MNQ9-17	Tornado Multiple Missile Protection of Cooling Towers	3

Procedures

Number	Description or Title	Revision
EN-LI-100	Process Applicability Determination	9
EN-LI-100	Process Applicability Determination	16
EN-LI-101	10 CFR 50.59 Evaluations	12
EN-DC-115	Engineering Change Process	17
EN-DC-117	Post Modification Testing and Special Instructions	7
EN-DC-136	Temporary Modifications	11
EN-DC-141	Design Inputs	15
EN-DC-213	Engineering Quality Review	6
OP-009-005	Shutdown Cooling	35
OP-500-008	Control Room Cabinet H	39
STA-O01-005	Leakage Testing of Air and Nitrogen Accumulators for Safety Related Valves	316
OP-903-121	Safety Systems Quarterly 1ST Valve Tests	19
OP-902-000	Standard Post Trip Actions	15
OP-902-001	Reactor Trip Recovery Procedure	14
OP-902-002	Loss of Coolant Accident Recovery	19
OP-902-003	Loss of Offsite Power/Loss of Forced circulation Recover	9
OP-902-004	Excess Steam Demand Recovery	15
OP-902-005	Station Blackout Recovery Procedure	18
OP-903-030	Safety Injection Pump Operability Verification	21
OP-903-108	SI Flow Balance Test	9

Procedures

Number	Description or Title	Revision
UNT-007-060	Control of Loose Items	304
EP-002-100	Technical Support Center (TSC) Activation, Operation, and Deactivation	42
EN-FAP-EP-010	Severe Weather Response	1

Drawings

Number	Description or Title	Revision
G-167 Sht. 1	Flow diagram – Safety Injection System	49
G-167 Sht. 2	Flow diagram – Safety Injection System	51
G-167 Sht.2	Safety Injection System	52
G-167 Sht. 3	Flow diagram – Safety Injection System	20
G-167 Sht. 4	Flow diagram – Safety Injection System	17
G-176 Sht. 1	Main Steam and Feedwater Piping	22
ESSE-SI206	SI-405B Bonnet Equalization Line	0
ESSE-SI205	SI-405A Bonnet Equalization Line	0
A223-6022 Sht.1	Booster Relay Support Rigid Restraint FWRR-6022	0
A223-6022 Sht.2	Booster Relay Support Rigid Restraint FWRR-6022	0
8469-123	FWST-2 Snubber Support	0

Work Orders

52478207 00372861 108105-03