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

August 8, 2017

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 INTEGRATED INSPECTION REPORT 05000382/2017002

Dear Mr. Chisum:

On June 30, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Waterford Steam Electric Station, Unit 3. On July 13, 2017, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented five findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

Further, inspectors documented a licensee-identified violation, which was determined to be of very low safety significance, in this report. The NRC is treating this violation as a NCV consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these NCVs, 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; and the NRC resident inspector at the Waterford Steam Electric Station, Unit 3.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; and the NRC resident inspector at the Waterford Steam Electric Station, Unit 3.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Geoffrey Miller, Branch Chief
Projects Branch D
Division of Reactor Projects

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

Enclosure: Inspection Report 05000382/2017002
w/ Attachments:
1. Supplemental Information
2. Information Request for the Occupational Radiation Safety Inspection
3. Cyber Security Follow-up Document Request

WATERFORD STEAM ELECTRIC STATION, UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000382/2017002 – AUGUST 8, 2017

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■ SUNSI Review By: JDixon/dll | ADAMS: ■ Yes ☐ No | ☐ Non-Publicly Available ■ Publicly Available | ■ Non-Sensitive ☐ Sensitive | Keyword: NRC-002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000382
License:	NPF-38
Report:	05000382/2017002
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	17265 River Road Killona, LA 70057
Dates:	April 1 through June 30, 2017
Inspectors:	F. Ramírez, Senior Resident Inspector C. Speer, Resident Inspector B. Correll, Project Engineer S. Graves, Senior Reactor Inspector N. Greene, Ph.D., Health Physicist R. Kopriva, Senior Reactor Inspector J. O'Donnell, CHP, Health Physicist
Approved By:	Geoffrey Miller Chief, Projects Branch D Division of Reactor Projects

SUMMARY

IR 05000382/2017002; 04/01/2017 – 06/30/2017; Waterford S.E.S., Unit 3; Adverse Weather Protection, Maint. Risk Assess. & Emergent Work Control, Op. Determinations & Func. Assess., Post Maint. Testing, Follow-up of Events & Notices of Enforce. Discretion

The inspection activities described in this report were performed between April 1 and June 30, 2017, by the resident inspectors at Waterford Steam Electric Station, Unit 3, and inspectors from the NRC's Region IV office. Five findings of very low safety significance (Green) are documented in this report. All of these findings involved violations of NRC requirements. Additionally, NRC inspectors documented in this report one licensee-identified violation of very low safety significance. The significance of inspection findings is indicated by their color (i.e., Green, greater than Green, White, Yellow, or Red), determined using Inspection Manual Chapter 0609, "Significance Determination Process," dated April 29, 2015. Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas," dated December 4, 2014. Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," dated July 2016.

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to assure that testing required to demonstrate that structures, systems, and components will perform satisfactorily while in service was identified and performed in accordance with written test procedures incorporating the requirements and acceptance limits contained in the applicable design documents. Specifically, prior to performing Licensee Procedure OP-903-027, "Inspection of Containment," Attachment 10.3, "Trisodium Phosphate Storage Basket Inspection," the licensee routinely performed a preliminary check to fill the trisodium phosphate storage baskets, thereby ensuring the successful completion of the technical specification-required surveillance. As a result, following unsatisfactory preliminary checks, the trisodium phosphate storage baskets were not evaluated for past operability. The licensee entered this condition into their corrective action program as Condition Report CR-WF3-2017-05108. The licensee's corrective actions will include performing the surveillance procedure as an as-found check and evaluating failed surveillances for past operability.

  The performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected its objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, conducting preliminary checks of the trisodium phosphate storage baskets and refilling them prior to performing the technical specification surveillance can mask the as-found condition of the test and preclude an evaluation of past operability if the levels are below the technical specification-required values. The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, "Significance Determination Process." Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," instructed the inspectors to use Appendix G, "Shutdown Operations Significance Determination Process." Using Appendix G, Attachment 1, Exhibit 3, "Mitigating Systems Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because the finding: (1) was not a deficiency affecting the design or qualification of a mitigating structure, system, or component; (2) did not represent a loss of system safety function; (3) did not represent an actual loss of safety function of at least a single train for greater than its technical specification allowed outage time or two separate safety systems out-of-service for greater than its technical specification allowed outage time; (4) with the cavity flooded, it did not represent an actual loss of safety function of one or more nontechnical specification trains of equipment during shutdown designated as risk-significant, for greater than 24 hours; (5) did not degrade the reactor coolant system level indication and/or core exit thermal couples when the cavity was not flooded; (6) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event; (7) did not involve fire brigade training and qualification requirements, or brigade staffing; (8) did not involve the response time of the fire brigade to a fire, and; (9) did not involve fire extinguishers, fire hoses, or fire hose stations.

The finding had a change management cross-cutting aspect in the area of human performance because leaders did not use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority. Specifically, when the licensee implemented the preliminary check practice in 2012, they did not evaluate the unintended consequences of how that practice would impact the results of the technical specification surveillance. Additionally, the licensee performed the preliminary check during each successive refueling outage between 2012 and 2017 giving the licensee an opportunity to identify the improper practice. As a result, the inspectors concluded this performance deficiency was indicative of current performance [H.3]. (Section 1R15)

- Green. The inspectors identified a non-cited violation of Technical Specification 6.8, "Procedures and Programs," and Regulatory Guide 1.33, "Quality Assurance Program Requirements," for the licensee's failure to perform operability testing on a safety-related component. Specifically, following the coil replacement of main steam isolation valve 2 solenoid valve, a safety-related component, the licensee did not perform a retest of the solenoid valve. As a result, main steam isolation valve 2 was returned to service without the assurance that no new deficiencies had been introduced, calling into question its operability. The licensee entered this condition into their corrective action program as Condition Report CR-WF3-2017-05507. The licensee's corrective action was to perform a voltage check of the solenoid valve to ensure it would energize in the event that a main steam isolation valve 2 closure was needed.

  The performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected its objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee restored main steam isolation valve 2 to an operable status without ensuring that its solenoid valve, which is a main steam isolation valve support system, was properly retested following maintenance. The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, "Significance Determination Process." Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," instructed the inspectors to use Appendix A, "Significance Determination Process for Findings At-Power." Using Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding to be of very low safety significance (Green) because it: (1) was not a deficiency affecting the design or qualification of a mitigating structure, system, or component; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for greater than its technical specification allowed outage time or two separate safety systems out-of-service for greater than its technical specification allowed outage time; and (4) did not represent an actual loss of function of one or more nontechnical specification trains of equipment designated as high safety-significant in accordance with licensee's maintenance rule program for greater than 24 hours.

The finding had a conservative bias cross-cutting aspect in the area of human performance because individuals did not use decision making-practices that emphasized prudent choices over those that were simply allowable. Specifically, the licensee did not make a conservative decision when determining whether the main steam isolation valve or its solenoid valve should be tested prior to proceeding with plant startup [H.14]. (Section 1R19)

- Green. The inspectors reviewed a self-revealed, non-cited violation of Technical Specification 6.8, "Procedures and Programs," and Regulatory Guide 1.33, "Quality Assurance Program Requirements," which occurred due to the licensee's failure to perform field work on reactor coolant loop 2 shutdown cooling warm-up valve, SI-135A. Specifically, mechanical maintenance technicians, who were assigned work on safety injection train A, erroneously performed work on safety injection train B on reactor coolant loop 1 shutdown cooling warm-up valve, SI-135B. As a result, both trains of emergency core cooling systems were simultaneously inoperable, which placed the plant in a 1-hour technical specification shutdown action statement. The licensee entered this condition into their corrective action program as Condition Report CR-WF3-2017-01433. The licensee's corrective actions included a revision of the model work order to require concurrent verification for component identification, and adding the valves to the protected equipment list for when the opposite train is inoperable.

  The performance deficiency was more than minor because it was associated with the configuration control attribute of the Mitigating Systems Cornerstone and adversely affected its objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, when the mechanics worked on valve SI-135B instead of valve SI-135A, they simultaneously made both trains of emergency core cooling systems inoperable. The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, "Significance Determination Process." Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," instructed the inspectors to use Appendix A, "Significance Determination Process for Findings At-Power." Using Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding to be of very low safety significance (Green) because it: (1) was not a deficiency affecting the design or qualification of a mitigating structure, system, and component; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for greater than its technical specification allowed outage time or two separate safety systems out-of-service for greater than its technical specification allowed outage time; and (4) did not represent an actual loss of function of one or more nontechnical specification trains of equipment designated as high safety-significant in accordance with licensee's maintenance rule program for greater than 24 hours.

  The finding had an avoid complacency cross-cutting aspect in the area of human performance because individuals did not recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes, and did not implement appropriate error reduction tools. Specifically, maintenance technicians repeatedly visited the incorrect work location and didn't properly verify the valve number to ensure they would work on the correct component [H.12]. (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. The inspectors identified multiple examples of a non-cited violation of Technical Specification 6.8, "Procedures and Programs," and Regulatory Guide 1.33, "Quality Assurance Program Requirements," for the licensee's failure to follow Licensee Procedure OP-901-521, "Severe Weather and Flooding," Revision 323. Specifically, on three occasions, the licensee did not close exterior doors when required by the procedure due to potential severe weather conditions. As a result, plant equipment was at an increased failure risk due to severe weather at the site. The licensee entered this condition into their corrective action program as Condition Reports CR-WF3-2017-03961 and CR-WF3-2017-04944. The licensee is planning corrective actions to ensure doors do not remain blocked open during conditions that require their closure.

  The performance deficiency was more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and adversely affected its objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to maintain all of the doors required by Licensee Procedure OP-901-521 with all fuel offloaded to the spent fuel pool threatened the licensee's ability to maintain the functionality of the spent fuel pool cooling system. The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, "Significance Determination Process," and determined that a qualitative analysis by a senior reactor analyst was required. The senior reactor analyst determined that the finding was of very low safety significance (Green). Using Inspection Manual Chapter 0609, Appendix M, "Signifiance Determination Process Using Qualitative Criteria," the senior reactor analyst performed a bounding analysis indicated that the total increase in core damage frequency from the failure to close the doors during severe weather was less than 1E-6.

  The finding had a work management cross-cutting aspect in the area of human performance because the organization did not implement a process of planning, controlling, and executing work activities such that nuclear safety was the overriding priority and the work process did not include the identification and management of risk commensurate to the work and the need for coordination with different groups of job activities. Specifically, during the planning and executing of work activities associated with Refueling Outage 21, the licensee did not consider the nuclear safety implications of blocking open exterior watertight and tornado doors and the work process did not include the identification and management of the risk associated with the blocked-open doors [H.5]. (Section 1R01)

- Green. The inspectors reviewed a self-revealed, non-cited violation of Technical Specification 6.8, "Procedures and Programs," and Regulatory Guide 1.33, "Quality Assurance Program Requirements," which occurred because the licensee did not implement instructions for maintaining containment integrity. Specifically, on April 18, 2017, the licensee did not ensure that the containment equipment hatch could be closed within the calculated reactor coolant system time to boil as required by Licensee Procedure OP-010-006, "Outage Operations," Revision 330. The licensee entered this condition into their corrective action program as Condition Report CR-WF3-2017-02541. The licensee's corrective actions included exiting the applicable condition, re-performing the equipment hatch closure drill to show the equipment hatch could be closed prior to the reactor coolant system time to boil, and performing repairs to the containment equipment hatch.

The performance deficiency was more than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone and adversely affected its objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee must close containment penetrations prior to the reactor coolant system time to boil in order to minimize radionuclide releases under accident conditions. The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, "Significance Determination Process." Inspection Manual Chapter 0609, instructed the inspectors to use Appendix H, "Containment Integrity Significance Determination Process," the inspectors determined the finding to be of very low safety significance (Green) because licensee maintained in-depth shutdown capability and because the duration of the performance deficiency was less than 8 hours.

The inspectors concluded that the finding had a teamwork cross-cutting aspect in the area of human performance because individuals and work groups did not communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained. Specifically, personnel performed work resulting in a short calculated reactor coolant system time to boil without first communicating their actions to operations or the outage control center, resulting in an unexpected plant condition [H.4]. (Section 1R13)

Licensee-Identified Violations

A violation of very low safety significance that was identified by the licensee has 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.

PLANT STATUS

The Waterford Steam Electric Station, Unit 3, began the inspection period at 100 percent power. The licensee initiated a plant shutdown on April 14, 2017, to begin Refueling Outage 21. On June 1, 2017, operators commenced a reactor startup. On June 13, 2017, the unit achieved full power. The unit remained at full power for the remainder of the inspection period.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

On June 23, 2017, the inspectors completed an inspection of the station's readiness for seasonal extreme weather conditions. The inspectors reviewed the licensee's adverse weather procedures for hurricane season and evaluated the licensee's implementation of these procedures. The inspectors verified that prior to hurricane season, the licensee had corrected weather-related equipment deficiencies identified during the previous hurricane season.

The inspectors selected two risk-significant systems that were required to be protected from hurricane season:

- Offsite electrical power
- Ultimate heat sink

The inspectors reviewed the licensee's procedures and design information to ensure the systems would remain functional when challenged by hurricane weather. The inspectors verified that operator actions described in the licensee's procedures were adequate to maintain readiness of these systems. The inspectors walked down portions of these systems to verify the physical condition of the adverse weather protection features.

These activities constituted one sample of readiness for seasonal adverse weather, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

.2 Readiness to Cope with External Flooding

a. Inspection Scope

On May 3, 2017, the inspectors completed an inspection of the station's readiness to cope with external flooding. After reviewing the licensee's flooding analysis, the inspectors chose three plant areas that were susceptible to flooding:

- Dry cooling tower areas
- Reactor auxiliary building
- Turbine building

The inspectors reviewed plant design features and licensee procedures for coping with flooding. The inspectors walked down the selected areas to inspect the design features, including the material condition of seals, drains, and flood barriers. The inspectors evaluated whether credited operator actions could be successfully accomplished.

These activities constituted one sample of readiness to cope with external flooding, as defined in Inspection Procedure 71111.01.

b. Findings

Introduction. The inspectors identified multiple examples of a Green, non-cited violation of Technical Specification 6.8, "Procedures and Programs," and Regulatory Guide 1.33, "Quality Assurance Program Requirements" for the licensee's failure to follow Licensee Procedure OP-901-521, "Severe Weather and Flooding," Revision 323. Specifically, on three occasions, the licensee did not close exterior doors when required by the procedure due to potential severe weather conditions. As a result, plant equipment was at an increased risk for failure due to severe weather at the site.

Description. The Licensee's Off-normal Procedure OP-901-521, "Severe Weather and Flooding," Revision 323, requires the licensee to take actions to prepare the site for the impact of potential severe weather. Included in these actions are to close the doors listed in Attachment 3, "Nuclear Island Exterior Watertight Doors Below Elevation +30 Ft MSL," during National Weather Service (NWS) issued flash flood watches or warnings, severe thunderstorm watches and warnings, or tornado watches or warnings. The procedure also directs licensee personnel to close the doors listed in Attachment 5, "Nuclear Island Exterior Tornado Doors," during NWS-issued tornado watches and warnings.

The inspectors determined that on three occasions, the licensee did not fully implement the actions required by Licensee Procedure OP-901-521:

- On April 30, 2017, during a tornado warning issued by the NWS for the site area

- On May 3, 2017, during a flash flood watch and a severe thunderstorm warning issued by the NWS for the site area

- On May 12, 2017, during a severe thunderstorm warning and tornado warning issued by the NWS for the site area

In support of Refueling Outage 21 activities, the licensee blocked open Doors 27, 33, and 36. Licensee Procedure OP-901-521, identifies Doors 27 and 33 as exterior watertight doors in Attachment 3, and Door 36 as an exterior tornado door in Attachment 5. Further, the licensee built temporary structures that prevented the closure of Doors 27 and 33. During the listed NWS-issued severe weather conditions, the licensee did not close the doors as required by Licensee Procedure OP-901-521.

Attachment 5 of Licensee Procedure OP-901-521, also specifies that, although not a tornado door, Door 33 should be closed when completing the attachment.

The reactor was defueled as part of Refueling Outage 21 during the severe weather conditions. The licensee completed offloading the core on April 26, 2017, and began reloading the core on May 13, 2017. The inspectors noted that during the planning for Refueling Outage 21, the licensee did not evaluate the impact that blocking the doors open or that constructing temporary structures would have on the site's ability to cope with severe weather conditions. Further, the licensee did not consider the impact of leaving the doors open until questioned by the inspectors following the severe weather conditions.

Analysis. The inspectors concluded that the failure to prepare the site for potential severe weather as required by Licensee Procedure OP-901-521, "Severe Weather and Flooding," Revision 323, was a performance deficiency which was reasonably within the licensee's ability to foresee and correct. The performance deficiency was more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and adversely affected its objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to maintain all of the doors required by Licensee Procedure OP-901-521 with all fuel offloaded to the spent fuel pool threatened the licensee's ability to maintain the functionality of the spent fuel pool cooling system.

The inspectors used NRC Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," to evaluate the finding for its impact on the Barrier Integrity cornerstone. The initial screening directed the inspectors to use Appendix G, "Shutdown Operations Significance Determination Process," to determine the significance of the finding and directed the inspectors to Attachment 1. Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process, Phase 1 Initial Screening and Characterization of Findings," states, "This appendix is intended to be used when the plant is shutdown with at least one fuel bundle in the reactor and temperature and pressure are within the normal Residual Heat Removal (RHR)/Decay Heat Removal (DHR) conditions, otherwise return to IMC 0609, Attachment 4, 'Initial Characterization of Findings.'" Because the failures of the licensee to prepare the site for potential severe weather all occurred during a time that the entire reactor core was offloaded to the spent fuel pool, the inspectors returned to Attachment 4.

In accordance with Attachment 4, Table 3, "SDP Appendix Router," the inspectors were then directed to go to IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Appendix A, Section 5.0, "Screening," provides that spent fuel pool findings are to be addressed in Exhibit 3, "Barrier Integrity Screening Questions." Given that this issue only affected the fuel in the spent fuel pool and spent fuel pool cooling systems, the inspectors screened the finding using Exhibit 3. Exhibit 3, Section D, Question 1 asks, "Does the finding adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis?" The inspectors determined that inadvertent flooding through the subject doorways would have resulted in a long-term loss of spent fuel pool cooling. This in turn would have resulted in the spent fuel pool exceeding the licensing basis temperatures. Therefore, the inspectors were directed to IMC 0609, Appendix M, "Significance Determination Process using Qualitative Criteria."

In accordance with Appendix M, Step 4.1, “Initial Bounding Evaluation,” the senior reactor analyst evaluated the direct impact of the performance deficiency on the risk to the fuel in the spent fuel pool. As part of a qualitative assessment, the analyst reviewed the licensee’s flood hazard evaluation, Areva Engineering Information Report 51-9227040-000, “Waterford Steam Electric Station Flooding Hazard Re-Evaluation Report.” Table 4-1, “Flood Evaluation Comparison,” provides a local intense precipitation flood height of 20.5 feet Mean Sea Level (MSL). The inspectors provided that the lowest entry point for the subject doorways was above 21 feet MSL. Therefore, by realistic assessment, a local flooding event, such as a flash flood in the area of Waterford 3, would not result in direct flooding of the plant areas of concern to the spent fuel pool. The analyst also performed a quantitative evaluation of the potential impact from tornados during the exposure period. According to the current analysis from the Office of Nuclear Reactor Research, the frequency of a tornado impacting Waterford 3 with a wind velocity great enough to accelerate significant missile strikes (>111 mph) is 2.41E-5/year. For the entire 56-day period of the outage, this would have resulted in a strike probability of 3.70E-6. Using a bounding analysis of the missile strike probability for two areas the size of the subject doors in open space, the analyst provided a missile strike probability of 9.65E-3. Therefore, a bounding probability of a tornado missile strike on the subject doors at any time during the outage was calculated to be 3.6E-8.

The bounding analysis indicated that the total increase in core damage frequency from the failure to close the doors during severe weather was less than 1E-6. In accordance with IMC 0609, Appendix M, “If the bounding evaluation shows that the finding is of very low safety significance (i.e., Green) the finding can be documented in accordance with IMC 0612 and the guidance provided in Step 4.3.” Step 4.3, “Process and Documentation,” states that a green finding should be documented in the inspection report, including the quantitative and/or qualitative method used including the results. Therefore, this finding is of very low safety significance (Green).

The finding had a work management cross-cutting aspect in the area of human performance because the organization did not implement a process of planning, controlling, and executing work activities such that nuclear safety was the overriding priority and the work process did not include the identification and management of risk commensurate to the work and the need for coordination with different groups of job activities. Specifically, during the planning and executing of work activities associated with Refueling Outage 21, the licensee did not consider the nuclear safety implications of blocking open exterior watertight and tornado doors and the work process did not include the identification and management of the risk associated with the blocked-open doors [H.5].

<u>Enforcement</u>. Technical Specification 6.8, “Procedures and Programs,” Section 1.a, requires, in part, that procedures shall be established, implemented and maintained covering, “the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2.” Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, Appendix A, Section 6.w, requires, in part, procedures for combatting, “Acts of Nature,” including tornados and floods. The licensee established Licensee Procedure OP-901-521, “Severe Weather and Flooding,” Revision 323, to meet the Regulatory Guide 1.33 requirement. Steps E1.4 of Licensee Procedure OP-901-521 requires, in part, that during NWS-issued severe thunderstorm or flash flood watches or warnings that the licensee verify, “all exterior doors and

hatches locked shut in accordance with Attachment 3." Step E2.5 of Licensee Procedure OP-901-521 requires, in part, that during NWS-issued tornado watches or warnings that the licensee verify, "all exterior doors and hatches locked shut in accordance with Attachment 3 and Attachment 5."

Contrary to the above, on April 30, May 3, and May 12, 2017, the licensee did not verify all exterior doors and hatches were locked shut during NWS-issued severe thunderstorm or flash flood watches or warnings, in accordance with Attachment 3 and Attachment 5 of Licensee Procedure OP-901-521. Specifically, during the NWS-issued flash flood watch on May 3, 2017, or the NWS-issued severe storm warnings on May 3 and May 12, 2017, the licensee did not verify Doors 27 or 33 were locked shut as required by Step E1.4 of Licensee Procedure OP-901-521. During the NWS-issued tornado warnings on April 30 and May 12, 2017, the licensee did not verify Doors 27, 33, or 36 were locked shut as required by Step E2.5 of Licensee Procedure OP-901-521. As a result, plant equipment was at an increased risk for failure due to severe weather at the site. The licensee entered this condition into their corrective action program as condition report CR-WF3-2017-03961 and CR-WF3-2017-04944. The licensee is planning corrective actions to ensure doors do not remain blocked open during conditions that require their closure.

Because this violation was of very low safety significance and the licensee entered the issue into their corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy.
(NCV 05000382/2017002-01, "Failure to Prepare the Site for Impending Adverse Weather")

1R04 Equipment Alignment (71111.04)

Partial Walk-Down

a. Inspection Scope

The inspectors performed partial system walk-downs of the following risk-significant systems:

- April 23, 2017, fuel pool cooling system due to high risk significance for system lineup
- May 10, 2017, emergency diesel generator train A with train B out of service for maintenance
- May 18, 2017, low pressure safety injection train B in shutdown cooling mode following re-alignment
- June 27, 2017, shield building ventilation train B with train A out of service for maintenance

The inspectors reviewed the licensee's procedures and system design information to determine the correct lineup for the systems. They visually verified that critical portions of the systems or trains were correctly aligned for the existing plant configuration.

These activities constituted four partial system walk-down samples, as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

The inspectors evaluated the licensee's fire protection program for operational status and material condition. The inspectors focused their inspection on six plant areas important to safety:

- April 17, 2017, containment building, Fire Area RCB-001
- April 24, 2017, fuel handling building, Fire Area FHB-001
- April 27, 2017, emergency diesel generator 3B, Fire Area RAB 15-001
- May 3, 2017, battery room 3A, Fire Area RAB 13-001
- May 3, 2017, battery room 3B, Fire Area RAB 11-001
- May 3, 2017, battery room 3AB, Fire Area RAB 12-001

For each area, the inspectors evaluated the fire plan against defined hazards and defense-in-depth features in the licensee's fire protection program. The inspectors evaluated control of transient combustibles and ignition sources, fire detection and suppression systems, manual firefighting equipment and capability, passive fire protection features, and compensatory measures for degraded conditions.

These activities constituted six quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On June 22, 2017, the inspectors completed an inspection of the station's ability to mitigate flooding due to internal causes. After reviewing the licensee's flooding analysis, the inspectors chose two plant areas containing risk-significant structures, systems, and components that were susceptible to flooding:

- Safeguards pump room A
- Safeguards pump room B

The inspectors reviewed plant design features and licensee procedures for coping with internal flooding. The inspectors walked down the selected areas to inspect the design features, including the material condition of seals, drains, and flood barriers. The inspectors evaluated whether operator actions credited for flood mitigation could be successfully accomplished.

These activities constituted completion of one flood protection measures sample, as defined in Inspection Procedure 71111.06.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

On May 2, 2017, the inspectors completed an inspection of the readiness and availability of risk-significant heat exchangers. The inspectors observed the licensee's inspection of the component cooling water train A heat exchanger and the material condition of the heat exchanger internals. Additionally, the inspectors walked down the component cooling water train A heat exchanger to observe its performance and material condition and verified that the component cooling water train A heat exchanger was correctly categorized under the Maintenance Rule and was receiving the required maintenance.

These activities constituted completion of one heat sink performance annual review sample, as defined in Inspection Procedure 71111.07.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

The activities described in subsections 1 through 4 below constituted completion of one inservice inspection sample, as defined in Inspection Procedure 71111.08.

.1 Nondestructive Examination Activities and Welding Activities

a. Inspection Scope

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Feedwater System	Steam Generator No. 2 Nozzle to Shell Weld. Report No. ISI-MT-17-001. Work Order 433430.	Magnetic Particle
Reactor Coolant System	Reactor CRDM Pressure Housing Upper Weld at Location 75. Report No. ISI-PT-17-018. Component Id. 02-Z-75X1. Work Order 433694	Penetrant

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor CRDM Pressure Housing Upper Weld at Location 84. Report No. ISI-PT-17-019. Component Id. 02-Z-84X1. Work Order 433694	Penetrant
Reactor Coolant System	Reactor CRDM Pressure Housing Upper Weld at Location 85. Report No. ISI-PT-17-020. Component Id. 02-Z-85X1. Work Order 433694	Penetrant
Reactor Coolant System	Reactor CRDM Pressure Housing Upper Weld at Location 87. Report No. ISI-PT-17-021. Component Id. 02-Z-87XI. Work Order 433694	Penetrant
Reactor Coolant System	Pressurizer Safety Valve Weld Overlays, Nozzle to 8” x 6” Reducer Safe-End Weld at 135 degrees. Report No. 151-UT-I7-020. Work Order 444230.	Ultrasonic
Reactor Coolant System	Pressurizer Safety Valve Weld Overlays, Nozzle to 8” x 6” Reducer Safe-End Weld at 45 degrees. Report No. 151-UT-I7-021. Work Order 444230.	Ultrasonic
Reactor Coolant System	Pressurizer Safety Valve Weld Overlays, Nozzle to 8” x 6” Reducer Safe-End Weld at 45 degrees. Report No. 151-UT-I7-022. Work Order 444230.	Ultrasonic
Feedwater System	Steam Generator No. 2 Inner Radius Nozzle Weld. Report No. ISI-UT-17-O16. Work Order 434430.	Ultrasonic
Reactor Coolant System	Reactor Head Closure Studs No. 28-54. Report No. ISI-UT-17-008. Work Order 433413	Ultrasonic
Safety Injection System	Safety Injection System 12” Elbow to Pipe Weld, Component Id. 17-017. Exam Angles 0, 45, 60, and 70 degrees. Report No. ISI-UT-I7-023. Work Order 433719.	Ultrasonic
Reactor Coolant System	Reactor Vessel Bottom Head Dome to Peel Segment Torus.	Ultrasonic

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
	Report No. ISI-VE-17-039. Work Order 433208.	
Reactor Coolant System	Reactor Vessel Peel Segment to Pell Segment at 30 degrees. Report No. ISI-VE-17-040. Work Order 433208.	Ultrasonic
Reactor Coolant System	Reactor Vessel Peel Segment to Peel Segment at 150 degrees. Report No. ISI-VE-17-042. Work Order 433208.	Ultrasonic
Reactor Coolant System	Reactor Vessel Outlet Nozzle to Shell at 0 degrees. Report No. ISI-VE-17-047. Work Order 433209.	Ultrasonic
Reactor Coolant System	Reactor Vessel Inlet Nozzle to Shell at 60 degrees. Report No. ISI-VE-17-048. Work Order 433209.	Ultrasonic
Reactor Coolant System	Reactor Vessel Inlet Nozzle to Shell at 240 degrees. Report No. ISI-VE-17-051. Work Order 433209.	Ultrasonic
Reactor Coolant System	Reactor Vessel Upper Shell to Flange Weld. Report No. ISI-VE-17-046. Work Order 433208.	Ultrasonic
Reactor Coolant System	Reactor Vessel Closure Head Bare Metal Visual Exam, Control Rod Drive Mechanism Nozzles 1-87. Report No. ISI-VT-17-132. Work Order 433538	Visual
Reactor Coolant System	Reactor Vessel Outlet Nozzle Inner Radius at 0 degrees. Report No. ISI–VE-17-054. Work Order 433208	Visual
Reactor Coolant System	Reactor Vessel Inlet Nozzle Inner Radius at 120 degrees. Report No. ISI–VE-17-056. Work Order 433208	Visual
Reactor Coolant System	Reactor Vessel Snubber Lug at 60 degrees. Report No. ISI–VE-17-011. Work Order 433410	Visual

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor Vessel Core Stop Lug at 40 degrees. Report No. ISI–VE-17-017. Work Order 433410	Visual
Reactor Coolant System	Reactor Vessel Surveillance Capsule Holder at 104 degrees. Report No. ISI–VE-17-004. Work Order 433410	Visual
Reactor Coolant System	Reactor Vessel Flow Baffle. Report No. ISI–VE-17-025. Work Order 433410	Visual
Reactor Coolant System	Reactor Vessel Core Barrel Exterior, (Top Section of Flange and Alignment, Under Flange to Barrel and Outlet Nozzle, and Lower Section of Core Barrel). Report No. ISI–VE-17-003. Work Order 433410	Visual
Reactor Coolant System	Reactor Vessel Core Barrel Interior (Under Flange to Barrel, Outlet Nozzle Surfaces at 0 and 180 degrees, and Baffle Support Ledge and Guide Lugs at 70, 160, 250, and 340 degrees. Report No. ISI–VE-17-003. Work Order 433410	Visual

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor CRDM Pressure Housing Lower Weld at Location 75. Report No. ISI-PT-17-006. Component Id. 02-W-75XI. Work Order 433695	Penetrant
Reactor Coolant System	Reactor CRDM Pressure Housing Lower Weld at location 84. Report No. ISI-PT-17-007. Component Id. 02-W-84X1. Work Order 433695	Penetrant
Reactor Coolant System	Reactor CRDM Pressure Housing Lower Weld at Location 85. Report No. ISI-PT-17-008. Component Id. 02-W-85X1. Work Order 433695	Penetrant

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor CRDM Pressure Housing Lower Weld at Location 87. Report No. ISI-PT-17-009. Component Id. 02-W-87XI. Work Order 433695	Penetrant
Reactor Coolant System	Pressurizer Safety Valve. Report No. ISI-VE-17-O01, Part No. BW 09724. Work Order 463827	Radiograph
Reactor Coolant System	Reactor Vessel Peel Segment to Peel Segment at 210 degrees. Report No. ISI-VE-17-043. Work Order 433208.	Ultrasonic
Reactor Coolant System	Reactor Vessel Bottom Head Assembly to Lower Shell Circular Weld. Report No. ISI-VE-17-027. Work Order 433208.	Ultrasonic
Reactor Coolant System	Reactor Vessel Lower Shell Course Longitudinal Weld at 330 degrees. Report No. ISI-VE-17-032. Work Order 433208.	Ultrasonic
Reactor Coolant System	Reactor Vessel Middle Shell Course Longitudinal Weld at 90 degrees. Report No. ISI-VE-17-033. Work Order 433208.	Ultrasonic
Reactor Coolant System	Reactor Vessel Middle Shell to Upper Shell Circular Weld. Report No. ISI-VE-17-029. Work Order 433208.	Ultrasonic
Reactor Coolant System	Reactor Vessel Inlet Nozzle Inner Radius at 240 degrees. Report No. ISI–VE-17-058. Work Order 433208	Visual
Reactor Coolant System	Reactor Vessel Snubber Lug at 0 degrees. Report No. ISI–VE-17-010. Work Order 433410	Visual
Reactor Coolant System	Reactor Vessel Snubber Lug at 120 degrees. Report No. ISI–VE-17-012. Work Order 433410	Visual

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor Vessel Core Stop Lug at 130 degrees. Report No. ISI–VE-17-019. Work Order 433410	Visual
Reactor Coolant System	Reactor Vessel Surveillance Capsule Holder at 97 degrees. Report No. ISI–VE-17-005. Work Order 433410	Visual

During the review and observation of each examination, the inspectors observed whether activities were performed in accordance with the ASME Code requirements and applicable procedures. The inspectors also reviewed the qualifications of all nondestructive examination technicians performing the inspections to determine whether they were current.

The inspectors directly observed a portion of the following welding activities:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor Coolant Loop 1A Cold Leg Temperature Thermowell Replacement. Weld Map No. 392245-01-01. Weld No. FW-1. Work Order 392245-01.	Gas Tungsten Arc Welding
Emergency Diesel Generator	Emergency Diesel Generator B Fuel Oil Feed Tank Moisture Removal Modification. Engineering Change No. 67960 adds drain line 3EG1-75 to the existing 3EG1-8B piping. Work Order 451204.	Shielded Metal Arc Welding

The inspectors reviewed records for the following welding activities:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Safety Injection System	Replace Valve SI-303A, Safety Injection Tank 1A Leakage Drain Valve. Weld Map No. 435402-01-01. Work Order 435402.	Gas Tungsten Arc Welding

The inspectors reviewed whether the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also determined whether essential variables were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications.

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

In compliance with ASME Code Case N-729-1, the licensee performed a direct examination of the bare-metal surface of the entire outer surface of the reactor head, including essentially 100 percent of the intersection of each nozzle with the reactor head. The reactor head surfaces were examined for evidence of leakage by the visual presence of boron crystals. The examination encompassed a full 360 degrees around the circumference of each penetration and extended from the reactor head to the shroud plenum plate. The activity was performed using remote visual testing video inspection examination. The inspectors witnessed and reviewed the results of the licensee's bare metal visual inspection of the reactor vessel head penetrations to determine whether the licensee had identified any evidence of boric acid challenging the structural integrity of the reactor head components and attachments. The inspectors also verified that the required inspection coverage was achieved and limitations were properly recorded. Each penetration was well marked for accurate identification. The inspectors reviewed the certifications of the personnel performing the inspection to confirm that the examiners were certified to their respective VT-2 nondestructive examination method, and that they had completed at least 4 hours of additional training in detection of borated water leakage from components and the resulting boric acid corrosion of adjacent ferritic steel components.

The inspectors were informed by the licensee that after the shutdown to commence Refueling Outage 21, and after the reactor head insulation package was removed, a white powdery substance was discovered on the reactor head flange and several reactor head studs. This substance was dry and white in color and did not appear to be affixed to the reactor head flange and studs. The white substance covered approximately one-third of the flange and studs on the south side of the reactor head. The white substance was identified by sample analysis as Molybdates (a corrosion inhibitor), and was not boric acid. Additional samples were provided to chemistry which also tested positive for molybdates. The molybdates located on the reactor head were from the control element drive mechanism cooler leak (Condition Report CR-WF3-2015-09693) that occurred through the entire Operating Cycle 21. The water that made contact with the head boiled off since the reactor is well above 212 degrees Fahrenheit, leaving the molybdates. Condition Report CR-WF3-2015-09693 identified that the component cooling water leak existed since the fall of 2015. The component cooling water leak dripping down to the reactor head, hit the head insulation package and the majority of the leakage was directed toward the edge of the insulation package and onto the reactor studs. There was a little bit of leakage that actually made its way down some control element drive mechanisms to the reactor head. This leakage was visually inspected, and when necessary, was analyzed. No boric acid was identified.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's implementation of its boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walk-down as specified in Licensee Procedures: CEP-BAC-001, "Boric Acid Corrosion Control (BACC) Program Plan," Revision 1, EN-DC-319, "Boric Acid Corrosion Control Program (BACCP)," Revision 11, and Program Section SEP-BAC-WF3-001, "Waterford 3 Boric Acid Corrosion Control Program (BACCP)," Revision 1. The inspectors reviewed whether the visual inspections emphasized locations where boric acid leaks could cause degradation of safety significant components, and whether engineering evaluation used corrosion rates applicable to the affected components and properly assessed the effects of corrosion induced wastage on structural or pressure boundary integrity. The inspectors observed whether corrective actions taken were consistent with the ASME Code and 10 CFR Part 50, Appendix B requirements.

The inspectors reviewed 12 licensee boric acid evaluations where boric acid deposits were found on reactor coolant system piping components and other components.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed the steam generator tube eddy current examination scope and expansion criteria to determine whether these criteria met technical specification requirements, Electric Power Research Institute (EPRI) guidelines, and commitments made to the NRC. The inspectors also reviewed whether the eddy current examination inspection scope included areas of degradations that were known to represent potential eddy current test challenges such as the top of tube sheet, tube support plates, and U-bends. The inspectors confirmed that repairs were required at the time of the inspection.

Steam Generator Inspection

- The inspectors verified that the number and sizes of steam generator tube flaws/degradation identified were consistent with the licensee's previous outage operational assessment predictions.

- The inspectors verified that steam generator eddy current examination scope and expansion criteria met technical specification requirements.

- The inspectors verified that eddy current probes and equipment configurations used to acquire data from the steam generator tubes were qualified to detect the known/expected types of steam generator tube degradation in accordance with

Appendix H, "Performance Demonstration for Eddy Current Examination of EPRI Document 1013706."

The inspectors reviewed the licensee's identification of the tube degradation mechanisms and no new degradation mechanisms were identified.

Tube Repair

The inspectors verified that the licensee implemented repair methods which were consistent with the repair processes allowed in the plant technical specification requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service. The licensee repaired a total of 27 tubes. The following repairs were made:

- Steam Generator 1 – 3 tubes plugged
- Steam Generator 2 – 24 tubes plugged

Secondary Side Inspections

- The inspectors reviewed the secondary side inspection results and verified that there was no observed degradation.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed 28 condition reports which dealt with inservice inspection activities and found the corrective actions for inservice inspection issues were appropriate. From this review the inspectors concluded that the licensee has an appropriate threshold for entering inservice inspection issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry inservice inspection operating experience. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On April 11, 2017, the inspectors observed a simulator training evolution for plant cooldown and placement of the shutdown cooling system in service. The inspectors assessed the performance of the operators and the evaluators' critique of their performance.

These activities constituted completion of one quarterly licensed operator requalification program sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

.2 Review of Licensed Operator Performance

a. Inspection Scope

On April 14 and 15, 2017, the inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity due to performing a plant shutdown to begin Refueling Outage 21. The inspectors observed the operators' performance of the following activities:

- Reactor coolant system boration
- Control element assembly insertion
- Alarm response
- Valve and pump manipulations
- Crew briefs

In addition, the inspectors assessed the operators' adherence to plant procedures, including Licensee Procedure EN-OP-115, "Conduct of Operations," and other operations department policies.

These activities constituted completion of one quarterly licensed operator performance sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

Routine Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed two instances of degraded performance or condition of safety-significant structures, systems, and components (SSCs):

- April 6, 2017, low pressure safety injection
- May 23, 2017, auxiliary component cooling water

The inspectors reviewed the extent of condition of possible common cause SSC failures and evaluated the adequacy of the licensee's corrective actions. The inspectors reviewed the licensee's work practices to evaluate whether these may have played a role in the degradation of the SSCs. The inspectors assessed the licensee's characterization of the degradation in accordance with 10 CFR 50.65 (the Maintenance Rule), and verified that the licensee was appropriately tracking degraded performance and conditions in accordance with the Maintenance Rule.

These activities constituted completion of two maintenance effectiveness samples, as defined in Inspection Procedure 71111.12.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed two risk assessments performed by the licensee prior to changes in plant configuration and the risk management actions taken by the licensee in response to elevated risk:

- April 4, 2017, nonstandard lift of component cooling water pump A replacement motor
- April 26, 2017, planned Yellow outage risk due to electrical safety bus A outage

The inspectors verified that these risk assessments were performed timely and in accordance with the requirements of 10 CFR 50.65 (the Maintenance Rule) and plant procedures. The inspectors reviewed the accuracy and completeness of the licensee's risk assessments and verified that the licensee implemented appropriate risk management actions based on the results of the assessments.

The inspectors also observed portions of four emergent work activities that had the potential to affect the functional capability of mitigating systems or to impact barrier integrity:

- April 18, 2017, emergent work related to the containment maintenance hatch
- May 3, 2017, emergent outage schedule change due to unavailability of train B safety buses
- May 8, 2017, unexpected failure of dry cooling tower fan 6A
- June 11, 2017, emergent work on reactor trip circuit breaker 2 and extent of condition review on other reactor trip circuit breakers

The inspectors verified that the licensee appropriately developed and followed a work plan for these activities. The inspectors verified that the licensee took precautions to minimize the impact of the work activities on unaffected SSCs.

These activities constituted completion of six maintenance risk assessments and emergent work control inspection samples, as defined in Inspection Procedure 71111.13.

b. Findings

Introduction. The inspectors reviewed a self-revealed, Green, non-cited violation of Technical Specification 6.8, “Procedures and Programs,” and Regulatory Guide 1.33, “Quality Assurance Program Requirements,” which occurred due to the licensee’s failure to implement Licensee Procedure OP-010-006, “Outage Operations,” Revision 330. Specifically, on April 18, 2017, the licensee did not ensure that the containment equipment hatch could be closed within the calculated reactor coolant system (RCS) time to boil as required by the procedure.

Description. On April 15, 2017, the licensee shutdown the reactor to begin Refueling Outage 21. On April 18, 2017, at approximately 6:50 a.m., refuel floor personnel began planned work that required the disassembly of in-core instrumentation, which is part of the RCS boundary. However, the refueling floor personnel did not communicate to the shift manager or the outage control center when the disassembly work began The disassembly of the in-core instrumentation resulted in an RCS calculated time to boil of 17.5 minutes.

Per Licensee Procedure OP-010-006, “Outage Operations,” Revision 330, Step 3.2.9.2.1, the licensee is required to be able to close and secure the containment equipment hatch before the calculated RCS time to boil. The poor communications put the licensee in a condition where containment penetrations were open without first verifying the ability to close them prior to the calculated RCS time to boil.

At approximately 10:00 a.m., after the shift manager and outage control center personnel became aware of the in-core instrument disassembly, the licensee performed a drill to verify the ability to close the containment equipment hatch within the time required by Licensee Procedure OP-010-006. Following the drill, the inspectors concluded that the licensee did not demonstrate the ability to meet Licensee Procedure OP-010-006 closure time requirements based on observations they made during the drill. These difficulties included a long delay to find an equipment operator and shortcomings involving hatch movement and hatch bolting. The licensee concluded that the drill results did not demonstrate their ability to meet Licensee Procedure OP-010-006 requirement to close the hatch prior to the calculated RCS time to boil. In the licensee-critique following the drill, the personnel responsible for closing the equipment hatch identified several issues and areas for improvement to ensure a more timely closure.

Following the drill, the licensee took action to reassemble the in-core instrumentation to exit the conditions requiring an RCS calculated time to boil and the associated Licensee Procedure OP-010-006 hatch closure requirements. At approximately 2:09 p.m., the licensee performed another drill and successfully demonstrated the ability to close the equipment hatch in approximately 16 minutes and 19 seconds. However, when the licensee attempted to re-open the equipment hatch, the device necessary to laterally move the equipment hatch failed thereby preventing the equipment hatch from opening. On April 19, 2017, the failed device was replaced and the equipment hatch was again opened at approximately 2:07 p.m.; however, the licensee did not re-perform a hatch closure drill following the maintenance to the equipment hatch. The inspectors questioned the licensee’s decision to not re-perform a hatch closure drill given the previous failure to demonstrate the ability to meet the closure time requirements of

Licensee Procedure OP-010-006 and the repairs performed on the equipment hatch. Following inspector questioning, on April 26, 2017, the licensee re-performed an emergency equipment hatch closure drill and demonstrated the ability to close the equipment hatch in approximately 13 minutes.

Analysis. The inspectors concluded that the failure to ensure that the containment equipment hatch could be closed in less than the RCS time to boil as required by Licensee Procedure OP-010-006 was a performance deficiency which was reasonably within the licensee's ability to foresee and correct. The performance deficiency was more than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone and adversely affected its objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee must be able to close containment penetrations prior to the RCS time to boil in order to minimize radionuclide releases under accident conditions.

The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, "Significance Determination Process." Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," instructed the inspectors to use Appendix G, "Shutdown Operations Significance Determination Process." Exhibit 4, "Barrier Integrity Screening Questions," of Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," to Appendix G, directed the inspectors to screen the issue in accordance with Appendix H, "Containment Integrity Significance Determination Process." Using Appendix H, the inspectors determined the finding to be of very low safety significance (Green) because the performance deficiency did not affect the likelihood of core damage, the licensee maintained in-depth shutdown capability, and the duration of the performance deficiency was less than 8 hours.

The inspectors concluded that the finding had a teamwork cross-cutting aspect in the area of human performance because individuals and work groups did not communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained. Specifically, personnel performed work resulting in a short calculated RCS time to boil without first communicating their actions to operations or the outage control center, resulting in an unexpected plant condition [H.4].

Enforcement. Technical Specification 6.8, "Procedures and Programs," Section 1.a, requires, in part, that procedures shall be established, implemented and maintained covering, "the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2." Regulatory Guide 1.33, "Quality Assurance Program Requirements," Revision 2, Appendix A, Section 3.f(1), requires that instructions be established for "maintaining containment integrity." The licensee established Licensee Procedure OP-010-006, "Outage Operations," Revision 330, to meet the Regulatory Guide 1.33 requirement. Step 3.2.9.2.1 of Licensee Procedure OP-010-006 requires, in part, that the equipment hatch shall be capable of being closed and secured, "before the calculated time for the reactor coolant system to boil."

Contrary to the above, on April 18, 2017, the equipment hatch was not capable of being closed and secured before the calculated time for the RCS to boil as required by Licensee Procedure OP-010-006. Specifically, the licensee initially attempted and failed to close and secure the equipment hatch prior to the 17.5 minute calculated time to boil. The licensee entered this condition into their corrective action program as Condition Report CR-WF3-2017-02541. The licensee's corrective actions included exiting the applicable condition, re-performing the equipment hatch closure drill to show the equipment hatch could be closed prior to the RCS time to boil, and performing repairs to the containment equipment hatch.

Because this violation was of very low safety significance (Green) and the licensee entered the issue into their corrective action program, this violation is treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy. (NCV 05000382/2017002-02, "Failure to Ensure Containment Equipment Hatch Closure Prior to RCS Time to Boil")

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed four operability determinations and functionality assessments that the licensee performed for degraded or nonconforming SSCs:

- April 14, 2017, operability determination of containment cooling fans train B
- May 3, 2017, operability determination of trisodium phosphate baskets inside containment
- May 5, 2017, operability determination of high pressure safety injection pump AB
- May 15, 2017, functionality assessment of reactor coolant pump 1B

The inspectors reviewed the timeliness and technical adequacy of the licensee's evaluations. Where the licensee determined the degraded SSC to be operable or functional, the inspectors verified that the licensee's compensatory measures were appropriate to provide reasonable assurance of operability or functionality. The inspectors verified that the licensee had considered the effect of other degraded conditions on the operability or functionality of the degraded SSC.

These activities constituted completion of four operability and functionality review samples, as defined in Inspection Procedure 71111.15.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to assure that testing required to demonstrate that SSCs will perform satisfactorily while in service was identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in the applicable design documents. Specifically, prior to performing Licensee Procedure OP-903-027, "Inspection of Containment," Attachment 10.3, "Trisodium Phosphate Storage Basket Inspection," the licensee routinely performed a preliminary check to fill the trisodium phosphate (TSP) storage baskets, thereby ensuring the successful completion of the technical specification-required surveillance.

Description. While Waterford was in Refueling Outage 21, in order to demonstrate that emergency core cooling systems are operable for operation in Modes 1, 2, and 3, Technical Specification Surveillance Requirement 4.5.2.d.3 requires that every 18 months the licensee verifies that a minimum total of 380 cubic feet of granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets. The purpose of the TSP storage baskets in the containment basement is to minimize the possibility of corrosion cracking of certain metal components during operation of the emergency core cooling systems following a loss of coolant accident. The TSP provides this protection by dissolving in the emergency core cooling systems sump water and causing the final pH to be raised to greater than or equal to 7.0.

To ensure the completion of Technical Specification Surveillance Requirement 4.5.2.d.3, the licensee created Work Order 52686268. Task 01 required operations personnel to verify the TSP basket volume in accordance with Licensee Procedure OP-903-027, “Inspection of Containment,” Attachment 10.3, “Trisodium Phosphate Storage Basket Inspection.” Task 02, for that same work order, instructed operations personnel to perform a preliminary check of the TSP baskets by following the instructions from the Licensee Procedure OP-903-027. Task 02 instructions stated that this initial check was performed in the middle of the outage to ensure adequate time is available should repair or TSP replenishment be needed. It also stated that the final verification for technical specification credit is performed near the end of the outage.

The inspectors noted that the licensee instituted the practice of performing this preliminary check during Refueling Outage 18 in 2012. The inspectors also noted that this practice of pre-checking the baskets and adding TSP, if necessary, would alter and adjust the physical condition of the TSP volume inside the baskets prior to the testing required per technical specifications, which would ultimately ensure that the technical specifications test results were satisfactory. When the inspectors reviewed results of all preliminary checks and surveillance tests since 2012, they discovered that during Refueling Outage 18 in 2012, the preliminary check results were unsatisfactory because the TSP amount was 367 cubic feet. Since this volume was below the technical specification required value of 380 cubic feet, the licensee added TSP to the baskets, and towards the end of the refueling outage completed the technical specification surveillance requirement satisfactorily. The inspectors noted that even though the emergency core cooling systems remained inoperable (for Modes 1, 2, and 3) until the successful completion of Licensee Procedure OP-903-027, the licensee did not evaluate for past operability once the low TSP condition was identified.

Analysis. The inspectors concluded that the licensee’s failure to ensure that testing of safety-related components would demonstrate that the components would perform satisfactorily in service was a performance deficiency which was reasonably within the licensee’s ability to foresee and correct. The performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected its objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, conducting preliminary checks of the TSP baskets and refilling them prior to performing the technical specification surveillance can mask the as-found condition of the test and preclude an evaluation of past operability if the levels are below the technical specification required values.

The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, "Significance Determination Process." Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," instructed the inspectors to use Appendix G, "Shutdown Operations Significance Determination Process." Using Exhibit 3, "Mitigating Systems Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because the finding: (1) was not a deficiency affecting the design or qualification of a mitigating SSC; (2) did not represent a loss of system safety function; (3) did not represent an actual loss of safety function of at least a single train for greater than its technical specification allowed outage time or two separate safety systems out-of-service for greater than its technical specification allowed outage time; (4) with the cavity flooded, it did not represent an actual loss of safety function of one or more nontechnical specification trains during shutdown designated as risk-significant, for greater than 24 hours; (5) did not degrade reactor coolant system level indication and/or core exit thermal couples when the cavity was not flooded; (6) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event; (7) did not involve fire brigade training and qualification requirements, or brigade staffing; (8) did not involve the response time of the fire brigade to a fire; and (9) did not involve fire extinguishers, fire hoses, or fire hose stations.

The finding had a change management cross-cutting aspect in the area of human performance because leaders did not use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority. Specifically, when the licensee implemented the preliminary check practice in 2012, they did not evaluate the unintended consequences of how that practice would impact the results of the technical specification surveillance. Additionally, the licensee performed the preliminary check during each successive refueling outage between 2012 and 2017, giving the licensee an opportunity to identify the improper practice. As a result, the inspectors concluded this performance deficiency was indicative of current performance [H.3].

<u>Enforcement</u>. Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program be "established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents."

Contrary to the above, from December 2012 until May 2017, for quality related components, to which Appendix B applies, the licensee failed to assure that testing required to demonstrate that SSCs will perform satisfactorily while in service was identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in the applicable design documents. Specifically, prior to performing Licensee Procedure OP-903-027, "Inspection of Containment," Attachment 10.3, "Trisodium Phosphate Storage Basket Inspection," the licensee performed a preliminary check to fill the TSP baskets, thereby ensuring a successful completion of the technical specification required surveillance. This practice resulted in one instance where unsatisfactory preliminary checks were not evaluated for past operability. The licensee entered this condition into their corrective action program as Condition Report CR-WF3-2017-05108. The corrective action will include performing the surveillance procedure as an as-found check and evaluating failed surveillances for past operability.

Because this violation was of very low safety significance (Green) and the licensee entered the issue into their corrective action program, this violation is treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy. (NCV 05000382/2017002-03, "Failure to Ensure Appropriate Testing of TSP Baskets Inside Containment")

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed two permanent plant modifications that affected risk-significant SSCs:

- April 25, 2017, startup transformer A relays
- May 15, 2017, vital and instrument static uninterruptable power supply (SUPS) upgrade project

The inspectors reviewed the design and implementation of the modifications. The inspectors verified that work activities involved in implementing the modifications did not adversely impact operator actions that may be required in response to an emergency or other unplanned event. The inspectors verified that post-modification testing was adequate to establish the operability of the SSCs as modified.

One permanent plant modification sample was reviewed by inspectors from the Region IV office. This sample and associated finding will be documented in Section 1R18 of Inspection Report 2017010. This additional sample is not reflected in this report.

These activities constituted completion of two samples of permanent modifications, as defined in Inspection Procedure 71111.18.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed seven post-maintenance testing activities that affected risk-significant SSCs:

- April 28, 2017, shutdown cooling system loop 2 suction isolation upstream inside valve
- May 2, 2017, 3A safety bus undervoltage relays
- May 4, 2017, shutdown cooling system loop 2 suction isolation downstream inside valve
- May 6, 2017, containment atmospheric release exhaust header A

- May 17, 2017, SUPS A, electric control panel A (ECP A), and power distribution panel A (PDP A)

- May 17, 2017, SUPS A1

- June 2, 2017, main steam isolation valve B

The inspectors reviewed licensing- and design-basis documents for the SSCs and the maintenance and post-maintenance test procedures. The inspectors observed the performance of the post-maintenance tests to verify that the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected SSCs.

These activities constituted completion of seven post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of Technical Specification 6.8, "Procedures and Programs," and Regulatory Guide 1.33, "Quality Assurance Program Requirements," for the licensee's failure to perform operability testing on a safety-related component. Specifically, following the coil replacement of main steam isolation valve 2 solenoid valve, a safety-related component, the licensee did not perform a retest of the solenoid valve. As a result, main steam isolation valve 2 was returned to service without the assurance that no new deficiencies had been introduced, calling into question its operability.

Description. On June 1, 2017, Waterford was in Mode 3 in the process of completing Refueling Outage 21 and preparing the plant for startup. At 3:55 a.m., while the plant was at normal operating pressure and temperature, the licensee declared main steam isolation valve 2 inoperable to replace the coil in one of its solenoid valves. The main steam isolation valves, which isolate the steam generators from the remaining portion of the secondary system in the event of a loss of coolant accident, are opened and closed by controlling the flow of hydraulic fluid into and out of an actuator cylinder. In order for the main steam isolation valve to close, the dump valves, driven by solenoid valves, release the hydraulic fluid to a reservoir. The solenoid valve coil replacement was performed as emergent work to address a ground in the 125-Volt DC bus train A, which had been reported on May 31, 2017. The licensee created Work Order 477276-03 to determine the source of the ground, and subsequently replace the solenoid valve coil.

Following the solenoid valve coil replacement, the licensee declared the main steam isolation valve operable and proceeded with plant startup. The licensee evaluated the need to perform a stroke retest of the main steam isolation valve but concluded that since the work only replaced the solenoid coil and did not change affect the hydraulic system, the work would not impact the performance of main steam isolation valve 2. The licensee determined that the stroke of main steam isolation valve 2 for post maintenance testing could take place during the next refueling outage in January 2019. However, the inspectors noted that the licensee did not consider testing the solenoid valve following the maintenance and as a result did not ensure that no new deficiencies had been introduced in the system. Work Order 477276-03 did not provide instructions to retest or check the solenoid valve following its installation in the plant. The inspectors noted that because no post maintenance test was performed following the solenoid valve maintenance, the main steam isolation valve would have gone an entire operating cycle without assurance that it was operable.

Licensee Procedure EN-WM-107, "Post Maintenance Testing," Revision 5, states, that operability testing will be performed for safety-related and technical specification related components affected by the work scope of the work order package. It also states that a post maintenance test should be performed following maintenance and troubleshooting activities that might have affected proper functioning of the component or associated components. The inspectors noted that since the licensee did not perform an operability test of the solenoid valve, the main steam isolation valve operability was called into question. Based on the lack of a post maintenance test for its solenoid valve, the inspectors concluded that the licensee's decision to not perform a post maintenance test was not conservative. Following inspector discussions with the licensee and the Office of Nuclear Reactor Regulation, the licensee satisfactorily performed voltage checks of the main steam isolation valves solenoid valve on June 29, 2017, to ensure that the solenoid valve would energize in the event that a main steam isolation valve 2 closure was needed.

Analysis. The inspectors concluded that the failure to perform operability testing on a safety-related component that was affected by the work scope of a work order package was a performance deficiency which was reasonably within the licensee's ability to foresee and correct. The performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected its objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee restored main steam isolation valve 2 to an operable status without ensuring that its solenoid valve, which is a main steam isolation valve support system, was properly retested following maintenance. As a result, the operability of main steam isolation valve 2 was called into question.

The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, "Significance Determination Process." Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," instructed the inspectors to use Appendix A, "Significance Determination Process for Findings At-Power." Using Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding to be of very low safety significance (Green) because it: (1) was not a deficiency affecting the design or qualification of a mitigating SSC; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for greater than its technical specification allowed outage time or two separate safety systems out-of-service for greater than its technical specification allowed outage time; and (4) did not represent an actual loss of function of one or more nontechnical specification trains of equipment designated as high safety-significant in accordance with licensee's maintenance rule program for greater than 24 hours.

The finding had a conservative bias cross-cutting aspect in the area of human performance because individuals did not use decision making practices that emphasized prudent choices over those that were simply allowable. Specifically, the licensee did not make a conservative decision when determining whether the main steam isolation valve or its solenoid valve should be tested prior to proceeding with plant startup [H.14].

Enforcement. Technical Specification 6.8, “Procedures and Programs,” Section 1.a, requires, in part, that procedures shall be established, implemented, and maintained covering, “the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2.” Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, Appendix A, Section 9.a, requires, in part, that, “maintenance that can affect the performance of safety-related equipment be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances.” The licensee established Licensee Procedure EN-WM-107, “Post Maintenance Testing,” Revision 5, to meet the Regulatory Guide 1.33 requirement. Step 5.2[1] of Licensee Procedure EN-WM-107 requires that operability testing be performed for safety-related and technical specification related components affected by the work scope of the work order package.

Contrary to the above, on June 1, 2017, the licensee did not perform operability testing for safety-related and technical specification related components affected by the work scope of the work order package. Specifically, following the coil replacement of main steam isolation valve 2 solenoid valve, a safety-related component, the licensee did not perform a retest of the solenoid valve. As a result, main steam isolation valve 2 was returned to service without the assurance that no new deficiencies had been introduced, calling into question its operability. The licensee entered this condition into their corrective action program as Condition Report CR-WF3-2017-05507. The licensee’s corrective action was to perform a voltage check of the solenoid valve to ensure it would energize in the event that a main steam isolation valve closure was needed.

Because this violation was of very low safety significance (Green) and the licensee entered the issue into their corrective action program, this violation is treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy. (NCV 05000382/2017002-04, “Failure to Perform a Post Maintenance Test on a Main Steam Isolation Valve Solenoid Valve”).

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

During the station’s refueling outage that concluded on June 1, 2017, the inspectors evaluated the licensee’s outage activities. The inspectors verified that the licensee considered risk in developing and implementing the outage plan, appropriately managed personnel fatigue, and developed mitigation strategies for losses of key safety functions. This verification included the following:

- Review of the licensee’s outage plan prior to the outage
- Review and verification of the licensee’s fatigue management activities
- Monitoring of shut-down and cool-down activities
- Verification that the licensee maintained defense-in-depth during outage activities
- Observation and review of reduced-inventory and mid-loop activities
- Observation and review of fuel handling activities
- Monitoring of heat-up and startup activities

These activities constituted completion of one refueling outage sample, as defined in Inspection Procedure 71111.20.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed five risk-significant surveillance tests and reviewed test results to verify that these tests adequately demonstrated that the SSCs were capable of performing their safety functions:

In-service tests:

- April 21, 2017, low pressure safety injection train A flow balance test

Containment isolation valve surveillance tests:

- April 26, 2017, containment penetration 10, containment purge inlet inside annulus valve and containment purge inlet inside containment valve

Other surveillance tests:

- April 13, 2017, main steam safety valves lift test
- May 3, 2017, overload bypass testing of high pressure safety injection header B to reactor coolant loops 1A and 2A flow control valves
- May 23, 2017, emergency diesel generator train A integrated test

The inspectors verified that these tests met technical specification requirements, that the licensee performed the tests in accordance with their procedures, and that the results of the test satisfied appropriate acceptance criteria. The inspectors verified that the licensee restored the operability of the affected SSCs following testing.

These activities constituted completion of five surveillance testing inspection samples, as defined in Inspection Procedure 71111.22.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors observed an emergency preparedness drill on June 28, 2017, to verify the adequacy and capability of the licensee's assessment of drill performance. The inspectors reviewed the drill scenario, observed the drill from the control room simulator and emergency operations facility, and attended the post-drill critique. The inspectors verified that the licensee's emergency classifications, offsite notifications, and protective action recommendations were appropriate and timely. The inspectors verified that any emergency preparedness weaknesses were appropriately identified by the licensee in the post-drill critique and entered into the corrective action program for resolution.

These activities constituted completion of one emergency preparedness drill observation sample, as defined in Inspection Procedure 71114.06.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Public Radiation Safety and Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors evaluated the licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities. The inspectors assessed the licensee's implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures. During the inspection, the inspectors interviewed licensee personnel, walked down various areas in the plant, performed independent radiation dose rate measurements, and observed postings and physical controls. The inspectors reviewed licensee performance in the following areas:

- Radiological hazard assessment, including a review of the plant's radiological source terms and associated radiological hazards. The inspectors also reviewed the licensee's radiological survey program to determine whether radiological hazards were properly identified for routine and nonroutine activities and assessed for changes in plant operations.

- Instructions to workers, including radiation work permit requirements and restrictions, actions for electronic dosimeter alarms, changing radiological condition, and radioactive material container labeling.

- Contamination and radioactive material control, including release of potentially contaminated material from the radiologically controlled area, radiological survey performance, radiation instrument sensitivities, material control and release criteria, and control and accountability of sealed radioactive sources.

- Radiological hazards control and work coverage. During walk-downs of the facility and job performance observations, the inspectors evaluated ambient radiological conditions, radiological postings, adequacy of radiological controls, radiation protection job coverage, and contamination controls. The inspectors also evaluated dosimetry selection and placement as well as the use of dosimetry in areas with significant dose rate gradients. The inspectors examined the licensee's controls for items stored in the spent fuel pool and evaluated airborne radioactivity controls and monitoring.

- High radiation area and very high radiation area controls. During plant walk-downs, the inspectors verified the adequacy of posting and physical controls, including areas of the plant with the potential to become risk-significant high radiation areas.

- Radiation worker performance and radiation protection technician proficiency with respect to radiation protection work requirements. The inspectors determined if workers were aware of significant radiological conditions in their workplace, radiation work permit controls/limits in place, and electronic dosimeter dose and dose rate set points. The inspectors observed radiation protection technician job performance, including the performance of radiation surveys.

- Problem identification and resolution for radiological hazard assessment and exposure controls. The inspectors reviewed audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution.

These activities constituted completion of the seven required samples of radiological hazard assessment and exposure control program, as defined in Inspection Procedure 71124.01.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as reasonably achievable (ALARA). The inspectors performed this portion of the attachment during the refueling outage in order to directly observe the licensee's ALARA process activities including planning, implementation of radiological work controls, execution of work activities, and ALARA review of work-in-progress. During the inspection the inspectors interviewed licensee personnel, reviewed licensee documents, and evaluated licensee performance in the following areas:

- Implementation of ALARA and radiological work controls. The inspectors observed pre-job briefings, reviewed planned radiological administrative, operational, and engineering controls, and compared the planned controls to field activities.

- Radiation worker and radiation protection technician performance during work activities performed in radiation areas, airborne radioactivity areas, or high radiation areas.

- Problem identification and resolution for ALARA and radiological work controls. The inspectors reviewed audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution.

These activities constituted completion of three of the five required samples of occupational ALARA planning and controls program, as defined in Inspection Procedure 71124.02, and completes the inspection.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures (MS05)

a. Inspection Scope

For the period of April 1, 2016, through March 31, 2017, the inspectors reviewed licensee event reports (LERs), maintenance rule evaluations, and other records that could indicate whether safety system functional failures had occurred. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, and NUREG-1022, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73," Revision 3, to determine the accuracy of the data reported.

These activities constituted verification of the safety system functional failures performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index: Emergency AC Power Systems (MS06)

a. Inspection Scope

The inspectors reviewed the licensee's mitigating system performance index data for the period of April 1, 2016, through March 31, 2017, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the mitigating system performance index for emergency ac power systems, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index: High Pressure Injection Systems (MS07)

a. Inspection Scope

The inspectors reviewed the licensee's mitigating system performance index data for the period of April 1, 2016, through March 31,2017, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the mitigating system performance index for high pressure injection systems, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.4 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors verified that there were no unplanned exposures or losses of radiological control over locked high radiation areas and very high radiation areas during the period of January 1, 2016, to March 31, 2017. The inspectors reviewed a sample of radiologically controlled area exit transactions showing exposures greater than 100 millirem. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the occupational exposure control effectiveness performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.5 Radiological Effluent Technical Specifications (RETS)/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors reviewed corrective action program records for liquid and gaseous effluent releases, and leaks and spills, that occurred between January 1, 2016, and March 31, 2017, to verify the performance indicator data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the radiological effluent technical specifications (RETS)/offsite dose calculation manual (ODCM) radiological effluent occurrences performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

Throughout the inspection period, the inspectors performed daily reviews of items entered into the licensee’s corrective action program and periodically attended the licensee’s condition report screening meetings. The inspectors verified that licensee personnel were identifying problems at an appropriate threshold and entering these problems into the corrective action program for resolution. The inspectors verified that the licensee developed and implemented corrective actions commensurate with the significance of the problems identified. The inspectors also reviewed the licensee’s problem identification and resolution activities during the performance of the other inspection activities documented in this report.

b. Findings

No findings were identified.

.2 Semiannual Trend Review

a. Inspection Scope

The inspectors observed plant activities and reviewed the licensee’s corrective action program, performance indicators, protected equipment lists, on-line risk assessments, and other documentation to identify trends that might indicate the existence of a more significant safety issue. The inspectors verified that the licensee was taking corrective actions to address identified adverse trends.

These activities constituted completion of one semiannual trend review sample, as defined in Inspection Procedure 71152.

b. Observations and Assessments

The inspectors identified a trend involving deficient risk assessments. Specifically, the inspectors identified three instances where on-line risk assessment tools and risk mitigation procedures where not applied in accordance with site expectations. The examples are discussed below:

- CR-WF3-2017-01433 – On March 8, 2017, while performing planned maintenance on the emergency core cooling system train A, the licensee failed to protect all the components of the emergency core cooling system train B in accordance with Licensee Procedure EN-WM-104, “Online Risk Assessment,” Revision 15. This contributed to maintenance personnel inadvertently commencing work on the reactor coolant loop 1 shutdown cooling warm up valve, SI-135B, which caused both trains of emergency core cooling systems to be simultaneously inoperable. As a result, the site entered a 1-hour shutdown limiting condition for operation. A finding associated with this assessment is documented in Section 4OA3.2 of this report.

- CR-WF3-2017-01553 – On March 15, 2017, during an auxiliary feedwater pump work window during which the pump was unavailable, the licensee failed to protect emergency feedwater pump AB under the protected equipment program. If emergency feedwater pump AB were to become unavailable during the auxiliary feedwater pump work window, the on-line risk would have increased to Orange. Per Licensee Procedure EN-OP-119, “Protected Equipment Postings,” Revision 7, if the loss of the redundant component or system would result in a risk escalation to Orange or Red, the site is required to place protected equipment postings.

- CR-WF3-2017-02235 – On April 12, 2017, licensee personnel were performing switchyard work that met the criteria for ‘heavy work’ as described in Licensee Procedure OI-037-000, “Operations Risk Assessment Guidelines,” Revision 310. Instead, the licensee categorized the switchyard work as ‘light work’. In addition, due to poor communications among multiple site departments, the equipment out of service database was not updated to reflect the ongoing switchyard work.

The inspectors discussed this trend with the licensee and ensured that each instance was captured in the corrective action program. The licensee entered the adverse trend in the corrective action program as Condition Report CR-WF3-2017-06457. The inspectors also ensured that the licensee’s planned and completed corrective actions would address and correct the condition.

c. Findings

No findings were identified.

.3 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors selected two issues for an in-depth follow-up:

- On May 19, 2017, the inspectors completed a review of a licensee operating experience evaluation documented in Condition Report CR-WF3-2013-01348. The licensee's evaluation was documented to review the applicability to Waterford of industry-wide issues associated with wedge pin failures in Anchor Darling motor operated double disk gate valves. The inspectors also reviewed Condition Report CR-HQN-2017-00655, where the licensee documented additional actions to evaluate recent developments associated with this issue.

  The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews, and compensatory actions. The inspectors verified that the licensee appropriately prioritized the planned corrective actions and that these actions were adequate to correct the condition.

- May 1–4, 2017, during an in-office inspection, the inspector reviewed the licensee-identified cyber security-related findings documented in Inspection Report 05000382/2015406, "Inspection of Implementation of Interim Cyber Security Milestones 1-7," for in-depth follow-up review. The inspector reviewed a sample of updated program documents and procedures, updated critical digital asset listings, training documents, and corrective action documents.

  The inspectors assessed the licensee's completed corrective actions. The inspectors verified that the licensee appropriately prioritized the corrective actions and that these actions were adequate to correct the conditions.

These activities constituted completion of two annual follow-up samples, as defined in Inspection Procedure 71152.

b. Findings

No findings were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Licensee Event Report (LER) 05000382/2015-007-01, "Both Emergency Diesel Generators Declared Inoperable"

a. Inspection Scope

This LER described additional information to what was initially contained in LER 2015-007-00, issued on October 23, 2015. The original LER described the circumstances surrounding the inoperability of both emergency diesel generators and the licensee's subsequent entry into Technical Specification 3.8.1.1, Condition F, which required one emergency diesel generator to be restored to operable within 2 hours or the plant be in hot standby within 6 hours. The revision to the LER provides additional information regarding the inoperability of emergency diesel generators A and B, adds causal information for the failure of a solenoid associated with the inlet air damper for emergency diesel generator B, information about similar prior issues, and provides the licensee's assessment of the nuclear safety significance of the issue.

The original LER was reviewed by the resident inspectors in NRC Inspection Report 05000382/2016002 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16218A383). The results of the review are documented in Section 4OA3 of that report. The residents confirmed that the additional information provided in the LER revision did not represent additional performance deficiencies. This LER is closed.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report (LER) 05000382/2017-001-00, "Both Trains of Emergency Core Cooling System Inoperable due to Inadvertently Performing Maintenance on Train 'B' Resulting in Event or Condition that Could Have Prevented Fulfillment of a Safety Function"

a. Inspection Scope

On March 8, 2017, control room operators identified that low pressure safety injection train B was inoperable due to the reactor coolant loop 1 shutdown cooling warmup valve, SI-135B, being found open, which is not the required position. At the time of discovery, low pressure safety injection train A was inoperable for maintenance and the station was in compliance with Technical Specification 3.5.2, Action 'A', which requires that an inoperable low pressure safety injection train be restored within 7 days. The control room operators entered Technical Specification 3.5.2, Action 'C' due to both trains of emergency core cooling systems being inoperable. Action 'C' requires that with both low pressure safety injection trains inoperable, at least one train must be restored within 1 hour. Valve SI-135B was subsequently closed and tested to verify operability. The licensee reported this event to the NRC as en event that could have prevented the fultillment of a safety function. However, after a safety analysis, the licensee concluded that low pressure safety injection train B would have been able to fulfill its safety function. The inspectors evaluated this issue and documented their findings below. This LER is closed.

b. Findings

Introduction. The inspectors reviewed a self-revealed, Green, non-cited violation of Technical Specification 6.8, "Procedures and Programs," and Regulatory Guide 1.33, "Quality Assurance Program Requirements," which occurred due to the licensee's failure to perform field work on reactor coolant loop 2 shutdown cooling warm-up valve, SI-135A. Specifically, mechanical maintenance technicians who were assigned work on safety injection train A, erroneously performed work on safety injection train B on reactor coolant loop 1 shutdown cooling warm-up valve, SI-135B. As a result, both trains of emergency core cooling systems were simultaneously inoperable, which placed the plant in a 1-hour shutdown limiting condition for operation action statement.

Description. On March 8, 2017, licensee personnel were performing planned maintenance on low pressure safety injection train A. As a result, low pressure safety injection train A was declared inoperable in accordance with Technical Specification 3.5.2, "ECCS Subsystems – Modes 1, 2, and 3." The site entered Technical Specification 3.5.2, Action 'A', which requires that an inoperable low pressure safety injection train be restored within 7 days. At 4:27 p.m. that same day, while maintenance in train A was ongoing, control room operators discovered that reactor coolant loop 1 shutdown cooling warmup valve, SI-135B, was open when its required standby position was closed and declared low pressure safety injection train B inoperable. Since both trains of low pressure safety injection were simultaneously inoperable, the licensee entered Technical Specification 3.5.2, Action 'C' which required at least one train be restored to an operable status within 1 hour. The licensee subsequently closed and restored valve SI-135B to an operable status. Action 'C' of Technical Specification 3.5.2 was exited at 5:05 p.m., and the site remained in compliance with Action 'A.'

The licensee's investigation discovered that mechanical maintenance personnel were scheduled to perform planned maintenance on the reactor coolant loop 2 shutdown cooling warmup valve, SI-135A. Instead, they incorrectly commenced work on valve SI-135B. Valves SI-135A and B are located on the -35 feet elevation of the reactor auxiliary building, on opposite sides. That elevation normally contains train 'A' components on the west side of the building, and train 'B' components in the east side of the building. However, a few components such as valves SI-135A and B, reside in the opposite side of the building. As such, valve SI-135A is located in the east side of the building among other train B components.

In preparation for the planned work on safety injection train A, protected equipment signs were installed on safety injection components train B; however, the protected equipment database did not identify the inclusion of a barrier around valve SI-135B. When the technicians performed the job site pre-job walkdown, they went to the west side of the -35 reactor auxiliary building (the wrong location) because they believed that this was where they would find train A components. The technicians were not aware that valve SI-135A was actually located on the east side of the reactor auxiliary building. Once they arrived at the job site, they did not verify that they were at the correct location by checking the component tags hung on the valve against Work Order 121138-13, the work order written for work on valve SI-135A. The technicians also failed to notice that the scaffold tags (to access the valve) were written for valve SI-135B.

After the pre-job brief, the three mechanical maintenance technicians returned to the job site and upon arrival performed a job site review at the base of the scaffold; this was another opportunity to identify that they were at the wrong location. The technicians also determined that the component verification would occur after they were on the scaffold next to the valve. Once on the scaffold, the lead technician held the work order in hand and pointed to the component identification number in the precaution step, and then pointed toward the component label hanging from the valve. The second technician provided nonverbal concurrence. Neither technician recognized that they were at the wrong component and commenced valve work.

The licensee's adverse condition analysis determined that Work Order 121138 was not planned in accordance with Licensee Procedure EN-HU-102, "Human Performance Traps and Tools." Specifically, Licensee Procedure EN-HU-102, Attachment 9.2, “Worker Human Performance Tools,” states that concurrent verification should be used for actions that have a high potential to lead to such consequences as a loss of safety function. In addition, the analysis determined that the protected equipment database should have included valve SI-135B as part of the equipment where protective barriers would be placed during the train A planned work window. Licensee Procedure EN-WM-104, “On-Line Risk Management,” states that for medium and high integrated risk activities, if the determined risk mitigation actions call for ‘protect redundant/mitigating component,’ the opposite train components located in mixed system train areas are positively protected.

Analysis. The inspectors concluded that the failure to perform field work on reactor coolant loop 2 shutdown cooling warm-up valve SI-135A and instead performing maintenance on valve SI-135B was a performance deficiency which was reasonably within the licensee’s ability to foresee and correct. The performance deficiency was more than minor, and therefore a finding, because it was associated with the configuration control attribute of the Mitigating Systems Cornerstone and adversely affected its objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, when the mechanics worked on valve SI-135B, they simultaneously made both trains of emergency core cooling systems inoperable. As a result, the licensee entered a 1-hour technical specification shutdown action statement.

The inspectors screened the finding in accordance with NRC Inspection Manual Chapter 0609, “Significance Determination Process.” Inspection Manual Chapter 0609, Attachment 4, “Initial Characterization of Findings,” instructed the inspectors to use Appendix A, “Significance Determination Process for Findings At-Power.” Using Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the inspectors determined the finding to be of very low safety significance (Green) because it: (1) was not a deficiency affecting the design or qualification of a mitigating SSC; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for greater than its technical specification allowed outage time or two separate safety systems out-of-service for greater than its technical specification allowed outage time; and (4) did not represent an actual loss of function of one or more nontechnical specification trains of equipment designated as high safety-significant in accordance with licensee’s maintenance rule program for greater than 24 hours.

The finding had an avoid complacency cross-cutting aspect in the area of human performance because individuals did not recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes, and did not implement appropriate error reduction tools. Specifically, maintenance technicians repeatedly visited the incorrect work location and didn’t properly verify the valve number to ensure they would work on the correct component. [H.12].

Enforcement. Technical Specification 6.8, “Procedures and Programs,” Section 1.a, requires, in part, that procedures shall be established, implemented, and maintained covering, “the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2.” Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, Appendix A, Section 9.a, requires, in part, that, “maintenance that can affect the performance of safety-related equipment be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances." The licensee established Work Order 121138-13 with documented instructions to perform field work on reactor coolant loop 2 shutdown cooling warm-up valve SI-135A to meet the Regulatory Guide 1.33 requirement.

Contrary to the above, on March 8, 2017, the licensee failed to perform field work on reactor coolant loop 2 shutdown cooling warm-up valve SI-135A. Specifically, mechanical maintenance technicians who were assigned work on safety injection train A, erroneously performed work on reactor coolant loop 1 shutdown cooling warm-up valve SI-135B. As a result, both trains of emergency core cooling systems were simultaneously inoperable, which placed the plant in a 1-hour technical specification shutdown action statement. The licensee entered this condition into their corrective action program as Condition Report CR-WF3-2017-01433. The licensee's corrective actions included a revision of the model work order to require concurrent verification for component identification, and added the valves to the protected equipment database list for when the opposite train is inoperable.

Because this violation was of very low safety significance (Green) and the licensee entered the issue into their corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy. (NCV 05000382/2017002-05, "Failure to Perform Maintenance on the Correct Safety-Related Component")

These activities constituted completion of two event follow-up samples, as defined in Inspection Procedure 71153.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 28, 2017, the inspectors presented the radiation safety inspection results to Mr. M. Chisum, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On May 4, 2017, the inspectors presented the cyber security in-office inspection results to Mr. M. Chisum, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors did not review any proprietary information.

On May 12, 2017, the inspectors presented the inservice inspection results to Mr. M. Chisum, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On July 13, 2017, the resident inspectors presented the inspection results to Mr. M. Chisum, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

4OA7 Licensee-Identified Violations

The following licensee-identified violation of NRC requirements was determined to be of very low safety significance (Green) and met the NRC Enforcement Policy criteria for being dispositioned as a non-cited violation:

- Licensee Audit LO-WLO-2016-00037, "Bioassay Program," dated November 21, 2016, identified that during Refueling Outage 20, staff reviewing air sample and lapel air sampler results had not been identifying positive results. The audit revealed that two positive lapel air samples from Refueling Outage 20 had not been identified nor had estimated personnel exposures been calculated. In addition, the audit identified seven positive air sample results which had no documented estimated exposures. As a result, dose was not assigned to individuals exposed to airborne radioactivity. As a result of the audit findings, the licensee retroactively assigned dose to three individuals working the October 25, 2015, cavity drain job in the amount of 36 mrem committed effective dose equivalent (CEDE) and 700 mrem committed dose equivalent (CDE) to bone surfaces and to one individual working on a November 8, 2015, decontamination job in the amount of 33 mrem CEDE and 661 mrem CDE to bone surfaces.

  Title 10 CFR 20.1703 states, in part, the licensee shall implement and maintain a respiratory protection program that includes: (1) air sampling sufficient to identify the potential hazard and estimate doses, and (2) surveys and bioassays, as necessary, to evaluate actual intakes.

  Contrary to the above, on November 21, 2016, the licensee failed to implement and maintain their respiratory protection program to include air sampling sufficient to identify the potential hazard and estimate doses, and surveys and bioassays, as necessary to evaluate actual intakes. Specifically, for two jobs and four individuals, the licensee failed to identify positive air sample results and assign internal dose to the subject individuals.

  In accordance with Inspection Manual Chapter 0612, Appendix B, "Issue Screening," the inspectors determined that the performance deficiency was more than minor. The finding adversely affected the Occupational Radiation Safety Cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation, in that, the failure to adequately assess internal exposure affects the licensee's ability to control and limit radiation exposure to the worker. Using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the inspectors determined that the finding was of very low safety significance (Green) because it did not involve: (1) as low as reasonably achievable (ALARA) planning and controls; (2) a radiological overexposure; (3) a substantial potential for an exposure; or (4) a compromised ability to assess the dose.

  The licensee's immediate corrective action was to coach all technicians on surveying airborne areas, ensure all air sample and lapel results were discussed with management, and count all air and lapel samples for alpha and beta to evaluate any potential internal radiation exposure. The licensee entered this issue into their corrective action program as Condition Report CR-WF3-2016-07300.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Allen, Nondestructive Examination Level III, Inspection Services
E. Braden, Senior Technician, Radiation Protection
D. Brenton, GMPO Staff, GMPO
J. Briley, Senior Nondestructive Examination Lead, Inspection Services
M. Briley, Fleet NDE Coordinator/Principal Level III, Corporate/Inspection Services
L. Brown, Licensing Specialist, Regulatory Assurance
J. Cary, Supervisor, Radiation Protection
M. Chisum, Site Vice President
D. James, Technician, Radiological Operations
J. Jarrell, Manager, Regulatory Assurance
B. Lanka, Engineering Director, Engineering
B. Lindsey, Operations Manager, Waterford 3 Operations
D. McLaren, Manager, Radiation Protection
M. McQueen, Senior Health Physics/Chemistry Specialist, Radiation Protection
C. Miller, Supervisor, Radiation Protection
M. Mills, Manager, NIDS
L. Milster, Regulatory Assurance, Licensing Engineer
R. O'Quinn, Senior Staff – Steam Generators, Engineering
R. Osborne, Unit Coordinator, Production
J. Rachal, Program Supervisor, Training
M. Rosen, Nondestructive Examination Supervior, Inspection Services
R. Sherman, ALARA Supervisor, Radiation Protection
D. Silia, Manager, Maintenance
L. Sire, Inservice Inspection Engineer, Engineering
M. Zamber, Senior Licensing Specialist, Regulatory Assurance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000382/2017002-01	NCV	Failure to Prepare the Site for Impending Adverse Weather (Section 1R01)
05000382/2017002-02	NCV	Failure to Ensure Containment Equipment Hatch Closure Prior to RCS Time to Boil (Section 1R13)
05000382/2017002-03	NCV	Failure to Ensure Appropriate Testing of TSP Baskets Inside Containment (Section 1R15)
05000382/2017002-04	NCV	Failure to Perform a Post Maintenance Test on a Main Steam Isolation Valve Solenoid Valve (Section 1R19)
05000382/2017002-05	NCV	Failure to Perform Maintenance on the Correct Safety-Related Component (Section 40A3)

Closed

05000382/2015007-01	LER	Both Emergency Diesel Generators Declared Inoperable (Section 4OA3)
05000382/2017001-00	LER	Both Trains of Emergency Core Cooling System Inoperable due to Inadvertently Performing Maintenance on Train ‘B’ Resulting in Event or Condition that Could Have Prevented Fulfillment of a Safety Function (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

Number	Title	Revision
EN-FAP-EP-010	Severe Weather Response	5
OP-901-521	Severe Weather And Flooding	323

Condition Reports (CRs)

CR-WF3-2017-03267	CR-WF3-2017-03445	CR-WF3-2017-03472	CR-WF3-2017-03961
CR-WF3-2017-04944	CR-WF3-2017-05242	CR-WF3-2017-05261	

Section 1R04: Equipment Alignment

Miscellaneous Documents

Number	Title	Revision/Date
G-167	Safety Injection System Flow Diagram	July 7, 1991
G-853	HVAC Air Flow Diagram Reactor Containment Building	23

Procedures

Number	Title	Revision
OP-002-006	Fuel Pool Cooling and Purification	216
OP-008-008	Shield Building Ventilation	10
OP-009-002	Emergency Diesel Generator	337
OP-009-005	Shutdown Cooling	38
OP-009-008	Safety Injection System	40

Condition Reports (CRs)

CR-WF3-2017-02764

Section 1R05: Fire Protection

Procedures

Number	Title	Revision
EN-DC-161	Control of Combustibles	17
FHB-001	Waterford-3 S.E.S Prefire Stratetgy Fuel Handling Building	9
FP-001-018	Pre-Fire Strategies, Development, and Revision	303
ME-003-006	Fire Barrier Penetration Seals	310
RAB 11-001	Waterford-3 S.E.S Prefire Stratetgy Elevation +21.00' RAB Battery Room "3B"	7
RAB 12-001	Waterford-3 S.E.S Prefire Stratetgy Elevation +21.00' RAB Battery Room "3AB"	7
RAB 13-001	Waterford-3 S.E.S Prefire Stratetgy Elevation +21.00' RAB Battery Room "3A"	7
RAB 15-001	Waterford-3 S.E.S Prefire Stratetgy Elevation +21.00' RAB (RCA) Emergency Diesel Generator "3B"	8
RCB-001	Waterford-3 S.E.S Prefire Stratetgy RCB General Area	11

Condition Reports (CRs)

CR-WF3-2017-02580 CR-WF3-2017-02603 CR-WF3-2017-03118

Section 1R06: Flood Protection Measures

Miscellaneous Documents

Number	Title	Revision
MNQ3-5	Flooding Analysis Outside Containment	5
PRA-W3-01-002	W3 Internal Flooding Analysis	3

Section 1R07: Heat Sink Performance

Miscellaneous Documents

Number	Title	Revision/Date
EPRI NP-7552	Heat Exhanger Performance Monitoring Guidelines	December 1991
SEP-HX-WF3-001	Generic Letter 89-13 Heat Exchanger Test Basis	1

Work Orders (WOs)

52586237

Section 1R08: Inservice Inspection Activities

Drawings

Number	Title	Revision
5817-11398	½ - 2 inch Carbon Steel Bolted Bonnet Globe Valve. Velan Drw. No.PI-76800-N 03	3
5817-11738 Sht. 1	Safety Injection Tank Leakage Drain Valves – 1 inch -1878 Socket End Stainless Steel, Non-Cobalt Trim Double Disc Gate Valve	0
5817-13747 Sht. 1 D-WC-11101-080	Closure Head Assembly	0
5817-13771 Sht. 1 RT-49641-R	Top Head Insulation System Key Layout	0
8469-3539 Sht. 1	Emergency Diesel – Piping Support Drawing for Support EGRR-5034	0
E-3029-LW3-EG-3	Emergency Diesel – Piping Isometric for Diesel Oil Day Tank “B” to Emergency Diesel Generator “B”	7
E-9270-163-004 Sht. 1-3	CEDM Installation Drawing	17
G 164 Sht. 1	Flow Diagram Miscellaneous Reactor Auxiliary Systems	47

Miscellaneous Documents

Number	Title	Revision/Date
1013706	EPRI – Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines, Final Report	October 2007
1014983	EPRI – Steam Generator In Situ Pressure Test Guidelines, Final Report	August 2017

Miscellaneous Documents

Number	Title	Revision/Date
1019038	EPRI – Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines, Final Report	November 2009
20170116-001	Nondestructive Evaluation (NDE), Summary of WESDYNE International, LLC Paragon to Paragon II Instrument Substitution for Piping and RPV Examination Procedures	January 16, 2017
20170223-001	Review of WesDyne Procedures No. PDI-ISI-254, Revision 8, PDI-ISI-254-NZ, Revision 2, and PDI-ISI-254-SI-NZ, Revision 1	February 24, 2017
EC 0000061743	Steam Generator Review of Prior Degradation Assessment and Operational Assessment per SG Integrity Assessment Guidelines	December 8, 2015
EC 0000070775 SG-SGMP-17-4	Generate a Degradation Assessment for RF21 Second ISI Inspection of the W3 Replacement SG per NEI 97-06 and EN-DC-317	March 30, 2017
EC 0000072048	Operability Input for Condition Report CR-WF3-2017-02567. Analysis of White Substance on Reactor Head	May 15, 2017
ECR-000015899	Document the Operational Assessment of SG Inspection Results From RF19 per NEI 97-06 and EN-DC-317	June 21, 2013
ECR-000020695	Waterford-3 Replacement Steam Generator Eddy Current RF21 ISI Probe Equivalency Report	0
LTR-SGMP-15-39	Waterford 3 SG Operational Assessment Review for Skip Cycle.	June 18, 2015
LO-WLO- 2012-00046	Licensee Self-Assessment: Snapshot Assessment of the Waterford 3 Welding and Section XI Repair/Replacement Programs	October 5, 2012
LO-WLO-2015-00065 CA-00001	Licensee Self-Assessment: Reactor Vessel Internal Inspection Focused Benchmark	March 30, 2016
LO-WLO-2016-00059	Self-Assessment Title: RF21WR3 Inservice Inspectoin Pre-NRC Inspection Snapshot Assessment	April 7, 2017
SEP-ISI-104 Table 3.3-1	Code Cases Incorporated into the ISI Program	5
SEP-ISI-104 Table 3.4-1	Requests for Relief and Requests for Alternatives from ASME Section XI Requirements	5
SG-SGMP-14-16	Waterford 3 Cycle 20 and 21 Steam Generator Operational Assessment	June 20, 2014
SG-SGMP-17-4	Waterford Unit 3 RE21 Outage Steam Generator Degradation Assessment	March 17, 2017

Miscellaneous Documents

Number	Title	Revision/Date
SGP-REP-INS-FP-GEN	Standard In-Situ Pressure Test Using the Computerized Data Acquistion System	4
WDI-PJF-1317482-TCR-002	Hartford Steam Boiler (ANI) Equivalency Letter for Paragon / Paragon II	February 10, 2016

Procedures

Number	Title	Revision
CEP-BAC-001	Boric Acid Corrosion Control (BACC) Program Plan	1
CEP-NDE-0255	Entergy Nuclear Engineering Programs, Radiographic Examination, ASME, ANSI, AWS, API, AWWA Welds and Components	8
CEP-NDE-0400	Entergy Nuclear Engineering Programs, Ultrasonic Examination	6
CEP-NDE-0404	Entergy Nuclear Engineering Programs, Manual Ultrasonic Examination of Ferritic Piping Welds	5
CEP-NDE-0423	Entergy Nuclear Engineering Programs, Manual Ultrasonic Examination of Austenitic Piping Welds (ASME XI)	7
CEP-NDE-0485	Entergy Nuclear Engineering Programs, Manual Ultrasonic Examination of Vessel Nozzle Inside Radius, (Non-App. VIII)	13
CEP-NDE-0504	Entergy Nuclear Engineering Programs, Ultrasonic Examination of Small Bore Diameter Piping for Thermal Fatigue Damage	4
CEP-NDE-0641	Entergy Nuclear Engineering Programs, Liquid Penetrant Examination (PT) for ASME Section XI	7
CEP-NDE-0731	Entergy Nuclear Engineering Programs, Magnetic Particle Examination (MT) for ASME Section XI	5
CEP-NDE-0901	Entergy Nuclear Engineering Programs, VT-1 Examination	4
CEP-NDE-0902	Entergy Nuclear Engineering Programs, VT-2 Examination	7
CEP-NDE-0903	Entergy Nuclear Engineering Programs, VT-3 Examination	5
CEP-NDE-0955	Entergy Nuclear Engineering Programs, Visual Examination (VE) of Bare-Metal Surfaces	303
CEP-NDE-0965	Entergy Nuclear Engineering Programs, Visual Welding Inspection ASME, ANSI B31.1	4
CEP-SG-003	Steam Generator Integrity Assessment	2

Procedures

Number	Title	Revision
CEP-WP-GWS-1	General Welding Standard ASME/ANSI – Entergy Nuclear Engineering Programs	2
EN-DC-319	Boric Acid Corrosion Control Program (BACCP)	11
FTK-ESPP-G00051	Boric Acid Corrosion Control Evaluations	5
SEP-BAC-WF3-OO1	Waterford 3 Boric Acid Corrosion Control Program (BACCP) Program Section	1
SEP-ISI-104	Program Section for ASME Section XI, Divisoin 1 Inservice Inspection Program	005
SEP-SG-WF3-001	Waterford 3 Steam Generator Program	0
SGP-REP-INS-FP-GEN	Standard Internal Review Sheet, Standard In-Situ Pressure Test Using the Computerized Data Acquistion System.	4
WDI-CAL-002	Standard Internal Review Sheet, Pulser/Receiver Linearity Procedure	11
WDI-PJF-1316965-EPP-001	2017 – Reactor Vessel 10-Year Examinations, Examination Program Plan (Scan Plan)	1
WDI-STD-005 (POI-ISI-254-NZ)	Standard Internal Review Sheet, Remote Inservice Inspection of Reactor Nozzle to Shell Welds	2
WDI-STD-088	Standard Internal Review Sheet, Underwater Remote Visual Examination of Reactor Vessel Internals	13
WDI-STD-1000 (PDI -ISI-254)	Standard Internal Review Sheet, Remote Inservice Inspection of Reactor Vessel Shell Welds	8
WDI-STD-1005	Standard Internal Review Sheet, Manual or Multi-Channel Automated Ultrasonic Instrument Linearity Procedure.	3

Relief Requests

Number	Title	Date
W3F1-2008-0013	Revision to Request for Alternative W3-ISI-005, Request to Use ASME Code Case N-716, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	February 14, 2008
W3F1-2011-0013	Request for Alternative W3-ISI-018, Inspection of Reactor Pressure Vessel Head Control Element Drive Mechanism Nozzles during the Third Ten-Year Inservice Inspection Interval, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	February 16, 2011

Relief Requests

Number	Title	Date
W3F1-2011-0014	Request for Alternative W3-ISI-019, Inspection of Reactor Vessel Head In-Core Instrument Nozzles during the Third Ten-Year Inservice Inspection Interval, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	February 16, 2011
W3F1-2011-0087	Commitment Change for Reactor Vessel Internals Degradation Management Program Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	December 19, 2011
W3F1-2012-0085	Waterford 3 Request for Alternative W3-ISI-020, ASME Code Case N-770-1 Baseline Examination Request for Alternative, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	October 16, 2012
W3F1-2012-0096	Waterford 3 Response to an NRC Request for Additional Information (RAI) associated with W3-ISI-020, Request for Alternative to ASME Code Case N-770-1 Baseline Examination [TAC No. ME9801], Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	November 15, 2012
W3F1-2012-0099	Waterford 3 Request for Alternative W3-ISI-021, ASME Code Case N-770-1 Baseline Examination Request for Alternative, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	November 30, 2012
W3F1-2012-0102	Waterford 3 Supplemental Response to an NRC Request for Additional Information (RAI) associated with W3-ISI-020, Request for Alternative to ASME Code Case N-770-1 Baseline Examination [TAC No. ME9801], Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	December 16, 2012
W3F1-2013-0044	Waterford 3 Request for Alternative W3-ISI-023, ASME Code Case N-770-1 Successive Examinations, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38	September 26, 2013

Condition Reports (CRs)

CR-WF3-2012-05939	CR-WF3-2014-01818	CR-WF3-2014-01826	CR-WF3-2014-01844
CR-WF3-2014-01861	CR-WF3-2014-01868	CR-WF3-2014-01917	CR-WF3-2014-01925
CR-WF3-2014-01932	CR-WF3-2014-01943	CR-WF3-2015-09693	CR-WF3-2016-00842
CR-WF3-2016-01257	CR-WF3-2016-01590	CR-WF3-2016-02552	CR-WF3-2016-02555
CR-WF3-2016-02787	CR-WF3-2016-03087	CR-WF3-2016-03101	CR-WF3-2016-03313
CR-WF3-2016-04234	CR-WF3-2016-04505	CR-WF3-2016-04694	CR-WF3-2016-07023

Condition Reports (CRs)

CR-WF3-2017-00239	CR-WF3-2017-00421	CR-WF3-2017-01225	CR-WF3-2017-02335
CR-WF3-2017-02423	CR-WF3-2017-02567	CR-WF3-2017-03089	CR-WF3-2017-03533
CR-WF3-2017-03833	CR-WF3-2017-03727	CR-WF3-2017-03834	CR-WF3-2017-03835
CR-WF3-2017-03836	CR-WF3-2017-05939	CR-WF3-2017-09693	

Work Orders (WOs)

380427	392245	433151	433208
433209	433410	433413	433430
433694	433695	433719	434430
434538	435402	444230	451204
470471			

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Procedures

Number	Title	Revision
EN-OP-115	Conduct of Operations	18
EN-OP-200	Plant Transient Response Rules	3
OP-010-005	Plant Shutdown	329

Section 1R12: Maintenance Effectiveness

Miscellaneous Documents

Number	Title	Revision/Date
	System Health Report ACC – Auxiliary Component Cooling Water	Q2-2015
	System Health Report ACC – Auxiliary Component Cooling Water	Q3-2015
	System Health Report ACC – Auxiliary Component Cooling Water	Q4-2015
	System Health Report ACC – Auxiliary Component Cooling Water	Q2-2016
	System Health Report ACC – Auxiliary Component Cooling Water	Q4-2016

Miscellaneous Documents

Number	Title	Revision/Date
	System Health Report SI – Safety Injection	Q1-2015
	System Health Report SI – Safety Injection	Q2-2015
	System Health Report SI – Safety Injection	Q3-2015
	System Health Report SI – Safety Injection	Q4-2015
	System Health Report SI – Safety Injection	Q2-2016
	System Health Report SI – Safety Injection	Q4-2016
EC 64530	SI Void Size	0

Procedures

Number	Title	Revision
EN-DC-203	Maintenance Rule Program	3
EN-DC-204	Maintenance Rule Scope and Basis	4
EN-DC-205	Maintenance Rule Monitoring	6
OP-903-026	Emergency Core Cooling System Valve Lineup Verification	26

Condition Reports (CRs)

CR-WF3-2015-03272	CR-WF3-2015-04076	CR-WF3-2016-04201	CR-WF3-2016-07481
CR-WF3-2017-03232	CR-WF3-2017-03253	CR-WF3-2017-03305	CR-WF3-2017-03324
CR-WF3-2017-03360	CR-WF3-2017-03449	CR-WF3-2017-04036	CR-WF3-2017-04081
CR-WF3-2017-04425			

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Miscellaneous Documents

Number	Title	Revision/Date
	Outage Risk Assessment Report	April 6, 2017
	Outage Schedule Change Reivew	May 2, 2017
EC 70578	Provide Full Core Offload Evaluation for Refuel 21 in Accordance With ECM98-67	0

Miscellaneous Documents

Number	Title	Revision/Date
EC 71961	Provide Input for Dry Cooing Tower Fan Requirments for Current Condition in Refuel 21. Defueled and Mode 6	0
ECM98-067	Limiting Single Failure Thermal-Hydraluic Analysis of Waterford 3 Spent Fuel Pool	1

Procedures

Number	Title	Revision
EN-MA-119	Material Handling Program	28
EN-OP-119	Protected Equipment Postings	8
EN-OU-108	Shutdown Safety Management Program	8
MM-008-001	Inside Maintenance Access Hatch and Outside Maintenance Access Hatch Shield Door Opening, Inspection, and Closing	12
OP-010-006	Outage Operations	330
OP-901-131	Shutdown Cooling Malfunction	34
PLG-009-014	Conduct of Planned Outages	315
UNT-007-008	Control of Heavy and Critical Loads	318
RF-001-013	Incore Insturment Flanges	311

Condition Reports (CRs)

CR-WF3-2017-02507	CR-WF3-2017-02533	CR-WF3-2017-02538	CR-WF3-2017-02541
CR-WF3-2017-02554	CR-WF3-2017-02568	CR-WF3-2017-02790	CR-WF3-2017-03185
CR-WF3-2017-05046	CR-WF3-2017-02825		

Work Orders (WOs)

00393027	00437206	00478120

Section 1R15: Operability Determinations and Functionality Assessments

Miscellaneous Documents

Number	Title	Revision/Date
EC-71373	Operability Input for SI Pump AB	April 20, 2017
EC-72022	RF21 Evaluation to Allow Loading Fuel with RCP 1B Snubber Removed	May 13, 2017
EC-72047	RCP Motor to Driver Mount Bolt Reduction for Mode 6	May 11, 2017

Miscellaneous Documents

Number	Title	Revision/Date
ECM05-003	High Pressrue Safety Injection System Capacity	1
ER-W3-99-0195-00-01	TSP Mitigation Outside Baskets At -11 RCB	March 18, 1999
LA170770-LR-001	Evaluation of As-Found Clearance on Bolt Heads at RCP Motor to Driver Mount Connection	May 14, 2017

Procedures

Number	Title	Revision
EN-OP-104	Operability Determination Process	11
OP-100-014	Technical Specification and Technical Requirements Compliance	338
OP-903-027	Inspection of Containment	304
OP-903-030	Safety Injection Pump Operability Verification	32
OP-903-037	Containment Cooling Fan Operability Verification	7

Condition Reports (CRs)

CR-WF3-2012-07140	CR-WF3-2014-02382	CR-WF3-2015-08661	CR-WF3-2017-02131
CR-WF3-2017-02315	CR-WF3-2017-02672	CR-WF3-2017-02853	CR-WF3-2017-03583
CR-WF3-2017-03973	CR-WF3-2017-05108		

Work Orders (WOs)

00153759	00237864	00335341	00338388
52367062	52486090	52582133	52686268

Section 1R18: Plant Modifications

Miscellaneous Documents

Number	Title	Revision
EC 43927	Vital and Instrument SUPS Upgrade Project	0
EC 58121	Engineering Input to CR-WF3-2015-03566	0
EC 63801	WF3 Fast Bus Transfer Supervisory Circuit Indication/Relay Replacment – ‘A’ Relays	0
EC 64897	Startup Transformer A Sudden Pressure Relay Modification	0

Procedures

Number	Title	Revision
EN-DC-115	Engineering Change Process	20
EN-DC-141	Design Inputs	15

Condition Reports (CRs)

CR-WF3-2015-03566

Work Orders (WOs)

00446539

Section 1R19: Post-Maintenance Testing

Drawings

Number	Title	Revision
B424 St1662	Steam Line 2 Isolation VA 2MS-V604B	22

Miscellaneous Documents

Number	Title	Date/Revision
	OCC Logs	June 1, 2017
CEP-IST-4	Standard on IST	308
ECT-43935-01	SUPS Phase II A Train – Test SUPS MA, Transfer A1 / MA, Transfer A1 / A, Test ECP MC / PDP MC	May 2, 2017
EOS 17-0357	Equipment Out of Srivce Checklist for MS-124B	June 1, 2017
SEP-WF3-IST-2	WF3 IST Plan	5

Procedures

Number	Title	Revision
EN-DC-117	Post Modification Testing and Special Instructions	9
EN-FAP-WM-002	Critical Evolutions	4
EN-WM-107	Post Maintenance Testing	5
ME-004-011	Limitorque Motor Operator Maintenance for SMB-0 Through SMB-4T Valves	301
ME-007-008	Motor Operated Valves	18
OP-903-033	Cold Shutdown IST Valve Tests	48

Procedures

Number	Title	Revision
OP-903-120	Containment and Miscellaneous Systems Quarterly IST Valve Tests	24
SEP-WF3-IST-1	WF3 Inservice Testing Bases Document	5

Condition Reports (CRs)

CR-WF3-2017-02368	CR-WF3-2017-04761	CR-WF3-2017-05159	CR-WF3-2017-05507

Work Orders (WOs)

00381051	00381400	00475224	00477276
52676974	52680949	52682372	52678838
52679220			

Section 1R20: Refueling and Other Outage Activities

Miscellaneous Documents

Number	Title	Revision/Date
	Outage Work Hours for 2 SROs, 2 NAOs, 2 Fire Brigade Member, 3 Maintenance Supervisor, 3 Maintenance Employee	May 12, 2017
EC 70578	Provide Full Core Offload Evaluation for Refuel 21 in Accordance With ECM98-067	0
ECM98-067	Limiting Single Failure Thermal-Hydraulic Analysis of Waterford 3 Spent Fuel Pool	1

Drawings

Number	Title	Revision/Date
G-167	Safety Injection System	19
5817-12131	Head, Extension, Fast Time Response RTD Assembly with QDC Connector	December 17, 1998
5817-13053	Thermowell, Primary Loop	October 11, 2000
5817-13055	Hot Leg Nozzles	January 23, 2000

Procedures

Number	Title	Revision
EN-OM-123	Fatigue Management Program	13
EN-OU-108	Shutdown Safety Management Program	8
MM-008-001	Inside Maintenance Access Hatch and Outside Maintenance Access Hatch Shield Door Opening, Inspection, and Closing	13
NOECP-256	Non Destructive Examination of Reactor Vessel Head, Upper Guide Structure and Core Support Barrel Lifting Rig Assemblies	2
OP-001-001	Reactor Coolant System Fill and Vent	34
OP-001-003	Reactor Coolant System Drain Down	318
OP-009-005	Shutdown Cooling	038
OP-009-008	Safety Injection System	40
OP-010-005	Plant Shutdown	329
OP-010-006	Outage Operations	329
OP-901-131	Shutdown Cooling Malfunction	34
PLG-009-014	Conduct of Planned Outages	315
PLG-009-018	Containment Coordination	3
RF-005-001	Fuel Movement	319

Condition Reports (CRs)

CR-WF3-2017-02356 CR-WF3-2017-02672 CR-WF3-2017-02754

Section 1R22: Surveillance Testing

Procedures

Number	Title	Revision
OP-901-312	Loss of Vital Instrument Bus	314
OP-903-100	MOV Overload Bypass Test	309
OP-903-108	SI Flow Balance Test	14
OP-903-115	Train A Integrated Emergency Diesel Geneartor/Engineering Safety Features Test	37
SEP-APJ-005	Waterford 3 Primary Containment Leakage Rate Testing (Appendix J) Program	6
STA-001-002	Containment Purge Valve Leakage Test	303

Procedures

Number	Title	Revision
STA-001-004	Local Leak Rate Test (LLRT)	314
STA-001-006	Leak Rate Testing	302
TD-C710.0045	Crosby Test Prcedure No. T-1652 for Determining Safety Valve Set Pressure with Air Set Pressure Device	1

Condition Reports (CRs)

CR-WF3-2017-02751	CR-WF3-2017-04384

Work Orders (WOs)

52680155	52680305	52683948	52685292

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Air Sample Surveys

Number	Title	Date
RF-04182017-020B	Insulation Removal +46 Pressurizer	April 18, 2017
RF-04202017-050	Blind Flange Support in Cavity	April 19, 2017
RF-04202017-051A	Flange Rotation in Lower Cavity	April 20, 2017
RF-04212017-070	Pressurizer Repack RC 301B +46 Elevation	April 21, 2017

Audits and Self-Assessments

Number	Title	Date
LO-WLO-2016-00015	Contamination Control	July 7, 2016
LO-WLO-2016-00037	Bioassay Program	January 16, 2017

Miscellaneous Documents

Number	Title	Date
	NSTS Annual Inventory Reconciliation Report	January 3, 2017

Miscellaneous Documents

Number	Title	Date
	Review of 10 CFR Part 61 Analyses	January 31, 2017
	Waterford 3 RF21 Daily Outage Report	April 24-28 2017
	Waterford 3 STM Alarm Setpoint Evaluation	March 24, 2016
Att. 9.6 to EN-RP-10	LHRA/VHRA Key Log	April 27, 2017
W/O 52665748	Semi-Annual Source Leak Test	July 6, 2016
W/O 5273448	Semi-Annual Source Leak Test	December 14, 2016

Procedures

Number	Title	Revision
EN-RE-220	PWR Control of Miscellaneous Material in the Spent Fuel Pool	03
EN-RP-100	Radiation Worker Expectations	11
EN-RP-101	Access Control for Radiologically Controlled Areas	12
EN-RP-102	Radiological Control	05
EN-RP-104	Personnel Contamination Events	09
EN-RP-108	Radiation Protection Posting	18
EN-RP-121	Radioactive Material Control	13
EN-RP-122	Alpha Monitoring	09
EN-RP-123	Radiological Controls for Highly Radioactive Objects	01
EN-RP-131	Air Sampling	15
EN-RP-143	Source Control	12
EN-RP-203	Dose Assessment	09
EN-RP-204	Special Monitoring Requirements	11
EN-RP-404	Operation and Maintenance of HEPA Vacuum Cleaners and HEPA Ventilation Units	06

Radiological Surveys

Number	Title	Date
WF3-1704-0207	Radwaste Solidification Building	April 11, 2017
WF3-1704-0613	-4 Reactor Containment Building	April 19, 2017
WF3-1704-0672	Monthly Surveillance of Low Level Radwaste Storage Building	April 20, 2017
WF3-1704-0744	-11 Reactor Containment Building	April 21, 2017
WF3-1704-0849	+46 Reactor Containment Building	April 23, 2017
WF3-1704-0853	+19.5 Shutdown Cooling	April 23, 2017
WF3-1704-0990	+21 Reactor Containment Building	April 25, 2017

Radiological Work Permits

Number	Title
20160118	Loading of Dry Fuel Storage Cask #18 with Spent Fuel and Transport
20170062	Entries into Posted Locked High Radiation Areas to Perform Minor Maintenance Activities, Walkdown, Surveillances and Inspections
20170110	Perform Work Activities in Alpha Level Three Areas in the Radiologically Controlled Area
20170509	RF21 Remove/Replace Steam Generator Primary Manways/Diaphragms
20170612	RF21 In-Service Inspection/FAC Testing and Dissimilar Metal (DM) Visual Exams in Containment

Condition Reports (CRs)

CR-WF3-2016-03660	CR-WF3-2016-03841	CR-WF3-2016-04425	CR-WF3-2016-05119
CR-WF3-2016-05796	CR-WF3-2016-05911	CR-WF3-2016-06066	CR-WF3-2016-06095
CR-WF3-2016-06573	CR-WF3-2016-07300	CR-WF3-2016-07480	CR-WF3-2017-00438
CR-WF3-2017-00553	CR-WF3-2017-02873	CR-WF3-2017-03046	

Section 2RS2: Occupational ALARA Planning and Controls

ALARA Planning, In-Progress Reviews, and Post-Job Reviews

Number	Title	Date
20160054	ALARA Plan	April 20, 2016
20160054	In Progress Review (Revision 6)	October 24, 2016

ALARA Planning, In-Progress Reviews, and Post-Job Reviews

Number	Title	Date
20160054	Post-Job Review	March 8, 2017
20170510	ALARA Plan	February 14, 2017
20170702	ALARA Plan	February 9, 2017
20170705	ALARA Plan	February 9, 2017

Audits and Self-Assessments

Number	Title	Date
LO-WLO-2016-00020	Focused Self-Assessment: Radiation Safety IP 71124.02 and 04 Pre-NRC Inspection	May 25, 2016

Miscellaneous Documents

Number	Title	Date
2016-2020	5 Year Exposure Reduction Plan	April 28, 2016
AMC-2016-08	ALARA Manager's Committee Meeting	November 1, 2016
RF2017-04 AMC	ALARA Manager's Committee Meeting	April 27, 2017

Pre-Approved Temporary Shielding Requests (TSR)

TSR	Tracking Number	Title	Date
RF-300	2012-57	Shield Pipe # 1SI3-214	October 11, 2016
RF-301	2015-21	+21 El., Across from Columns 16-18	September 29, 2016
RF-303	2002-25	Various High Rad Components in RCB	September 29, 2016
RF-304	2002-03	Shield (3) Seal Injection Filters	September 29, 2016

Procedures

Number	Title	Revision
EN-RP-105	Radiological Work Permits	16
EN-RP-109	Hot Spot Program	05
EN-RP-110	ALARA Program	14
EN-RP-110-04	Radiation Protection Risk Assessment Process	07
EN-RP-110-06	Outage Dose Estimating And Tracking	01
HP-001-114 0	Control Of Temporary Shielding	16
UNT-001-016 0	Radiation Protection	303

Radiation Work Permits

Number	Title	Revision
20160054	SI Tank Sampling, Personnel/Escape Interlock Door Tests, RCP 2A Speed Probe Troubleshooting/Repairs, Minor Maintenance, Inspections and Valve Lineups, RP Job Coverage into Posted LHRA's	6
20170510	Install/Remove Steam Generator Nozzle Dams, Pin Verification, and Closeout.	0
20170511	To Perform Eddy Current Work/Tube Plugging Inside of the Steam Generators Primary Side and Equipment Staging/Destaging.	0
20170702	RF21 Disassembly of Reactor Head and All Associated Work Activities.	1
20170705	RF 21 Reassembly of Reactor Head and Associated Work Activities Including Staging/Destaging of Equipment.	0

Condition Reports (CRs)

CR-WF3-2016-01908	CR-WF3-2016-01954	CR-WF3-2016-02500	CR-WF3-2016-02601
CR-WF3-2016-02834	CR-WF3-2016-04215	CR-WF3-2016-05797	CR-WF3-2016-06143
CR-WF3-2016-06846	CR-WF3-2016-06904	CR-WF3-2016-07036	CR-WF3-2016-07245
CR-WF3-2016-07443	CR-WF3-2017-00086	CR-WF3-2017-00888	

Section 4OA1: Performance Indicator Verification

Miscellaneous Documents

Number	Title	Revision/Date
ECH-NE-09-00036	Waterford 3 Mitigating Systems Performance Index Basis Document	6
LTR-SATH-17-013	Evaluation of Compromised LPSI System at Waterford 3	0
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	7
W3F1-2016-0050	NRC Performance Indicator (PI) Data – 2nd Quarter 2016	July 18, 2016
W3F1-2016-0052	NRC Performance Indicator (PI) Data – Change Report (CR) Emergency Preparedness	July 21, 2016
W3F1-2016-0068	NRC Performance Indicator (PI) Data – 3rd Quarter 2016	October 12, 2016
W3F1-2016-0076	NRC Performance Indicator (PI) Data – Change Report 3rd Quarter MSPI (INPO chg)	November 17, 2016
W3F1-2017-0007	NRC Performance Indicator (PI) Data – 4th Quarter 2016 (October, November and December)	January 12, 2017
W3F1-2017-0008	NRC Performance Indicator (PI) Data – Change Report Data 3rd Quarter 2016 Emergency Preparedness – Drill/exercise Performance	January 10, 2017
W3F1-2017-0036	NRC Performance Indicator (PI) Data – 1st Quarter 2017 ROP Data	October 12, 2016

Procedures

Number	Title	Revision
EN-LI-114	Regulatory Performance Indicator Process	7

Section 4OA2: Problem Identification and Resolution

Miscellaneous Documents

Number	Title	Revision/Date
	Network Diagram, PA Security and Video Level 4	
	Network Diagram, SOCA Security System Level 4	
	Network Diagram, SOCA Security and Video Level 4	
	Waterford 3 Plant Monitoring Computer w/ Data Diode and Business LAN Components	

Miscellaneous Documents

Number	Title	Revision/Date
EC-45714	Technical Manual Updates for TD-M120.0100 and TD-A391.0155	0
EC-48521	14” Flowserve & Anchor Darling Valves	2
OE Evaluation	Wedge Pin Failure in Anchor Darling Motor Operated Double Disk Gate Valves with Threaded Stem to Upper Wedge Connections	May 1, 2013
SFAQ 16-02	Deterministic Devices	January 24, 2017
SFAQ 16-05	Moving Data between Security Levels	March 7, 2017
TD-A391.0155	Anchor Darling Operating and Instruction Manual for Double Disk Gate Valve	1

Procedures

Number	Title	Revision
EN-FAP-IT-008	Nuclear Cyber Security Training and Awareness	4
EN-FAP-IT-105	Computer System Walkdowns	1
EN-IT-103	Nuclear Cyber Security Program	12
EN-IT-103-01	Control of Portable Digital Media Connected to Critical Digital Assets	11

Condition Reports (CRs)

CR-WF3-2013-01398	CR-WF3-2013-03893	CR-WF3-2015-01378	CR-WF3-2015-01593
CR-WF3-2015-04276	CR-WF3-2015-04277	CR-WF3-2015-04674	CR-WF3-2015-05026
CR-WF3-2015-05277	CR-WF3-2016-05916	CR-HQN-2017-00655	

Work Orders (WOs)

00004680	00009653	00009687

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Miscellaneous Documents

Number	Title	Revision
LTR-SATH-17-013	Evaluation of Compromised LPSI System at Waterford 3	A

Procedures

Number	Title	Revision
EN-FAP-OM-012	Prompt Investigations and Notifications	17
EN-LI-118	Cause Evaluation Process	24

Condition Reports (CRs)

CR-WF3-2017-01433 CR-WF3-2017-01518

Work Orders (WOs)

00121138

The following items are requested for the
Occupational Radiation Safety Inspection
Waterford-3
April 24-28, 2017
Integrated Report 2017002

Inspection areas are listed in the attachments below.

Please provide the requested information on or before **April 7, 2017**.

Please submit this information using the same lettering system as below. For example, all contacts and phone numbers for Inspection Procedure 71124.0**1** should be in a file/folder titled "**1**- A," applicable organization charts in file/folder "**1**- B," etc.

If information is placed on *ims.certrec.com*, please ensure the inspection exit date entered is at least 30 days later than the onsite inspection dates, so the inspectors will have access to the information while writing the report.

In addition to the corrective action document lists provided for each inspection procedure listed below, please provide updated lists of corrective action documents at the entrance meeting. The dates for these lists should range from the end dates of the original lists to the day of the entrance meeting.

If more than one inspection procedure is to be conducted and the information requests appear to be redundant, there is no need to provide duplicate copies. Enter a note explaining in which file the information can be found.

If you have any questions or comments, please contact Natasha Greene at (817) 200-1154 or Natasha.Greene@nrc.gov.

PAPERWORK REDUCTION ACT STATEMENT

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, control number 3150-0011.

1. Radiological Hazard Assessment and Exposure Controls (71124.01) and Performance Indicator Verification (71151)

Date of Last Inspection: **March 21, 2016**

A. List of contacts and telephone numbers for the Radiation Protection Organization Staff and Technicians

B. Applicable organization charts

C. Audits, self-assessments, and LERs written since date of last inspection related to this inspection area

D. Procedure indexes for the radiation protection procedures

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.

1. Radiation Protection Program Description
2. Radiation Protection Conduct of Operations
3. Personnel Dosimetry Program
4. Posting of Radiological Areas
5. High Radiation Area Controls
6. RCA Access Controls and Radiation Worker Instructions
7. Conduct of Radiological Surveys
8. Radioactive Source Inventory and Control
9. Declared Pregnant Worker Program

F. List of corrective action documents (including corporate and sub-tiered systems) since date of last inspection

a. Initiated by the radiation protection organization
b. Assigned to the radiation protection organization

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide in document formats which are "searchable" so that the inspector can perform word searches.

If not covered above, a summary of corrective action documents since date of last inspection involving unmonitored releases, unplanned releases, spills or releases in which any dose limit or administrative dose limit was exceeded (for Public Radiation Safety Performance Indicator verification in accordance with IP 71151)

G. List of radiologically significant work activities scheduled to be conducted during the inspection period (If the inspection is scheduled during an outage, please also include a list of work activities greater than 1 rem, scheduled during the outage with the dose estimate for the work activity.)

H. List of active radiation work permits

I. Radioactive source inventory list

   a. All radioactive sources that are required to be leak tested

   b. All radioactive sources that meet the 10 CFR Part 20, Appendix E, Category 2, and above threshold. Please indicate the radioisotope, initial and current activity (w/assay date), and storage location for each applicable source.

J. The last two leak test results for the radioactive sources inventoried and required to be leak tested. If applicable, specifically provide a list of all radioactive source(s) that have failed its leak test within the last two years

K. A current listing of any non-fuel items stored within your pools, and if available, their appropriate dose rates (Contact / @ 30cm)

L. Computer printout of radiological controlled area entries greater than 100 millirem since the previous inspection to the current inspection entrance date. The printout should include the date of entry, some form of worker identification, the radiation work permit used by the worker, dose accrued by the worker, and the electronic dosimeter dose alarm set-point used during the entry (for Occupational Radiation Safety Performance Indicator verification in accordance with IP 71151).

2. Occupational ALARA Planning and Controls (71124.02)
Date of Last Inspection: **September 12, 2016**

A. List of contacts and telephone numbers for ALARA program personnel

B. Applicable organization charts

C. Copies of audits, self-assessments, and LERs, written since date of last inspection, focusing on ALARA

D. Procedure index for ALARA Program

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.

1. ALARA Program
2. ALARA Committee
3. Radiation Work Permit Preparation

F. A summary list of corrective action documents (including corporate and sub-tiered systems) written since date of last inspection, related to the ALARA program. In addition to ALARA, the summary should also address Radiation Work Permit violations, Electronic Dosimeter Alarms, and RWP Dose Estimates

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide in document formats which are "searchable" so that the inspector can perform word searches.

G. List of work activities greater than 1 rem, since date of last inspection, Include original dose estimate and actual dose.

H. Site dose totals and 3-year rolling averages for the past 3 years (based on dose of record)

I. Outline of source term reduction strategy

J. If available, provide a copy of the ALARA outage report for the most recently completed outages for each unit

K. Please provide your most recent Annual ALARA Report.

Cyber Security Follow-up Document Request

NOTE: If any requested documents are identified as security-related, please notify the lead inspector:

Sam Graves
RIV/DRS/EB2
1600 E. Lamar Blvd.
Arlington, TX 76011

1. Corrective action documents for NRC- and Licensee-identified performance deficiencies described in the Milestones (MS) 1-7 Inspection Report (2015405). Please provide the plant documents that corrected the deficiencies (e.g., revised procedures, work orders, modification packages, new equipment, et cetera).

2. Current Cyber Security Program document(s)

3. Cyber Security program procedures

4. List of contacts with contact information

5. Cyber security group organization chart

6. Diagram of defensive network

7. A list of critical digital assets identified since the last onsite week of the MS 1-7 Inspection

8. A list of Cyber Security Program changes since the MS 1-7 Inspection