UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

August 13, 2013

Donna Jacobs, Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – NRC INTEGRATED INSPECTION REPORT 05000382/2013003

Dear Ms. Jacobs:

On June 30, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Waterford Steam Electric Station, Unit 3 facility. The enclosed inspection report documents the inspection results, which were discussed on July 11, 2013, with Mr. Carl Rich, Director, Nuclear Safety Assurance, and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Two NRC identified and one self-revealing findings of very low safety significance (Green) were identified during this inspection.

All of these findings were determined to involve violations of NRC requirements. Additionally, the NRC has determined that two traditional enforcement Severity Level IV violations occurred. The NRC is treating these violations as non-cited violations consistent with Section 2.3.2.a of the Enforcement Policy.

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

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 Regional Administrator, Region IV; and the NRC Resident Inspector at Waterford Steam Electric Station, Unit 3 facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Donald B. Allen, Chief
Project Branch E
Division of Reactor Projects

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

Enclosure: Inspection Report 05000382/2013003
w/ Attachment: Supplemental Information

cc w/ encl: Electronic Distribution

Electronic distribution by RIV:

Acting Regional Administrator (Steven.Reynolds@nrc.gov)
Acting Deputy Regional Administrator (Thomas.Bergman@nrc.gov)
DRP Director (Kriss.Kennedy@nrc.gov)
DRP Deputy Director (Troy.Pruett@nrc.gov)
DRS Director (Tom.Blount@nrc.gov)
DRS Deputy Director (Jeff.Clark@nrc.gov)
Senior Resident Inspector (Marlone.Davis@nrc.gov)
Resident Inspector (Mica.Baquera@nrc.gov)
WAT Administrative Assistant (Linda.Dufrene@nrc.gov)
Branch Chief, DRP/E (Don.Allen@nrc.gov)
Senior Project Engineer, DRP/E (Ray.Azua@nrc.gov)
Project Engineer, DRP/E (Jim.Melfi@nrc.gov)
Project Engineer, DRP/E (Dan.Bradley@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Public Affairs Officer (Lara.Uselding@nrc.gov)
Project Manager (Kaly.Kalyanam@nrc.gov)
Branch Chief, DRS/TSB (Ray.Kellar@nrc.gov)
RITS Coordinator (Marisa.Herrera@nrc.gov)
ACES (R4Enforcement.Resource@nrc.gov)
Regional Counsel (Karla.Fuller@nrc.gov)
Congressional Affairs Officer (Jenny.Weil@nrc.gov)
Technical Support Assistant (Loretta.Williams@nrc.gov)
RIV/ETA: OEDO (Vivian.Campbell@nrc.gov)

R:_REACTORS_WAT\2013\WAT2013003-RP-MD.docx
ACCESS NUMBER: ML13225A748

SUNSI Rev Compl.	☑Yes ☐ No	ADAMS	☑Yes ☐ No	Reviewer Initials	DBA
Publicly Avail.	☑Yes ☐ No	Sensitive	☐Yes ☑ No	Sens. Type Initials	DBA

SRI:DRP/E	RI:DRP/E	SPE:DRP/E	C:DRS/EB1	C:DRS/EB2	C:DRS/OB
MDavis	MBaquera	RAzua	TFarnholtz	GMiller	VGaddy
/RA/E	/RA/E	/RA/	/RA	/DLoveless for/	/RA/
8/13/13	8/13/13	8/12/13	8/7/13	8/7/13	8/8/13
C:DRS/PSB1	C:DRS/PSB2	C:DRS/TSB	BC:DRP/E		
MHaire	JDrake	RKellar	DAllen		
/RA/	/RA/	/TFarnholtz for/	/RA/		
8/8/13	8/8/13	8/7/13	8/13/13		

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000382
License:	NPF-38
Report:	05000382/2013003
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	17265 River Road Killona, LA 70057
Dates:	April 1 through June 30, 2013
Inspectors:	M. Davis, Senior Resident Inspector M. Baquera, Acting Senior Resident Inspector R. Azua, Senior Project Engineer D. Bradley, Project Engineer G. George, Senior Reactor Inspector M. Williams, Reactor Inspector N. Okonkwo, Reactor Inspector
Approved By:	Donald B. Allen, Chief Project Branch E Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000382/2013003; 04/01/2013 – 06/30/2013; Waterford Steam Electric Station, Unit 3, Integrated Resident & Regional Report; Adv. Weather Prot.; Eval of Changes, Test, or Exper & Perm Plant Mods; Refuel. & Other Outage Act.; PI&R; Other Activities.

The report covered a 3-month period of inspection by resident inspectors and an announced baseline inspection by region-based inspectors. Three Green non-cited violations of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The cross-cutting aspect is determined using Inspection Manual Chapter 0310, "Components Within the Cross-Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors reviewed a self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, because the licensee did not provide post modification testing instructions for activities affecting quality that were appropriate to the circumstances and that included appropriate acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee did not provide adequate post modification testing instructions for vibration monitoring of the feedwater piping system that included appropriate acceptance criteria following the installation of the new replacement steam generators. As a result, the plant experienced an automatic reactor trip and a subsequent down power due to an increase in vibrations on the feedwater piping system without appropriate acceptance criteria and monitoring during power ascension. The licensee entered this issue into their corrective action program as CR-WF3-2013-0445. The immediate corrective actions taken to restore compliance included the implementation of a revised vibration-monitoring plan to include appropriate acceptance criteria and the development of engineering changes to mitigate vibration effects on the feedwater piping system.

  The failure to provide adequate post modification testing instructions for vibration monitoring of feedwater piping system following steam generator replacement was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations. The inspectors used the NRC Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings" and Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," to evaluate this issue. The inspectors determined that the finding was of very low safety significance (Green) because the transient initiator did not contribute to the likelihood that mitigation equipment or functions would not be available. The inspectors concluded that the finding reflected current licensee performance and involved a cross-cutting aspect in the operating

experience component of the problem identification and resolution area in that the licensee did not implement operating experience through changes to station equipment to support plant safety [P.2.b] (Section 4OA5).

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, because the licensee did not provide design control measures for verifying or checking the adequacy of the features designed to withstand the effects of a probable maximum precipitation (PMP) flooding event on the reactor auxiliary building (RAB) roof areas. Specifically, the licensee did not provide an analysis to demonstrate that adequate flood protection existed from the effects of a PMP flooding event on safety-related components and electrical equipment located on the roof of the RAB in the main steam isolation valve (MSIV) wing areas. As a result, the licensee did not perform an analysis to determine if expected ponding levels from a PMP flooding event would challenge safety-related components and electrical equipment such as the emergency feedwater flow control and isolation valves and cables, main steam isolation valves and cables, atmospheric dump valves, and back-up nitrogen accumulator components. The licensee entered this issue into their corrective action program as CR-WF3-2012-7520. The immediate corrective actions taken to restore compliance included the performance of a preliminary analysis to show that the installed scuppers and roof drains have margin to protect against a local PMP flooding event and that the ponding depth would have little or no affect on the safety-related equipment and cables located in the MSIV wing areas.

  The failure to provide design control measures for verifying or checking the adequacy of the features designed to withstand the effects of a local PMP on the RAB roof areas was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating System cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the safety-related equipment located on the RAB roof in the MSIV wing areas are required to safely shutdown and maintain the reactor in a cold shutdown condition following accidents and anticipated operational occurrences. The inspectors used the NRC Inspection Manual Chapter 0609, Attachment 4, “Initial Characterization of Findings” and Appendix A, “The Significance Determination Process for Findings At-Power,” to evaluate this issue. The inspectors determined that the finding was of very low safety significance (Green) because it did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding, or severe weather initiating event. The inspectors concluded that the finding reflected current licensee performance and involved a cross-cutting aspect in the corrective action program component of the problem identification and resolution area in that the licensee did not identify potential flooding issues completely, accurately, and in a timely manner commensurate with their safety significance [P.1.a] (Section 1R01).

- Severity Level IV. The inspectors identified a non-cited violation of 10 CFR 50.73(a)(1) because the licensee did not submit a Licensee Event Report (LER) in a timely manner after the discovery of a reportable event. Specifically, the licensee failed to submit a required LER within 60 days after the discovery of a condition that

affected the manual hand-wheel operation of safety related air operated valves following a loss of their corresponding back-up nitrogen accumulators. The licensee determined that the manual hand-wheel function on the essential chiller and emergency feedwater isolation and backup flow control valves did not work. The licensee was aware of the condition that existed but did not adequately evaluate the condition as a part of their reportability review. The licensee entered this issue into their corrective action program as CR-WF3-2013-2564. The immediate corrective actions taken to restore compliance included a new reportability review of the condition and the development of an LER.

The failure to submit a required LER within 60 days after discovery of a condition that required a report was a violation of NRC requirements. The inspectors determined that this violation was also a performance deficiency. However, the inspectors determined that the performance deficiency was minor. The inspectors considered this issue to be within the traditional enforcement process because it had the potential to impact the NRC's ability to perform its regulatory oversight function. The inspectors used the NRC Enforcement Policy to evaluate the significance of this violation. The inspectors determined that the violation was a Severity Level IV because it was similar to an example provided in Section 6.9 of the NRC Enforcement Policy. The inspectors did not assign a cross-cutting aspect to this non-cited violation because there was no finding associated with this traditional enforcement violation (Section 4OA2).

Cornerstone: Barrier Integrity

- Severity Level IV. The inspectors identified a Severity Level IV non-cited violation for the licensee's failure to update the final (updated) safety analysis report in accordance with 10 CFR 50.71(e). Specifically, from July 1981 to April 18, 2013, the licensee failed to update the methodology, the data input, and the resulting limits for the fuel bundle drop accident analysis in the Waterford Steam Electric Station, Unit 3, Updated Final Safety Analysis Report (UFSAR), Section 15.7.3.4, "Design Basis Fuel Handling Accidents." This violation was entered into the licensee's corrective action program as Condition Report CR-WF3-2013-0193.

  The failure to update the methodology, the data input to the calculation, and the resulting limits for the fuel bundle drop accident analysis in Section 15.7.3.4 of the UFSAR in accordance with 10 CFR 50.71(e) is a performance deficiency. This performance deficiency was evaluated using traditional enforcement because it has the potential to impact the NRC's ability to perform its regulatory function. The inspectors used the NRC Enforcement Policy to evaluate the significance of this violation. Consistent with the NRC Enforcement Policy, the inspectors determined that the performance deficiency is a Severity Level IV non-cited violation. This non-cited violation had no cross-cutting aspect because there was no finding associated with this traditional enforcement violation (Section 1R17).

- Green. The inspectors identified a non-cited violation of Waterford Steam Electric Station, Unit 3, Technical Specification (TS) Limiting Condition of Operation (LCO) 3.3.1 because the licensee did not take action to suspend operations that involved reactivity changes to accomplish startup activities with only one excore nuclear instrumentation (ENI) logarithmic (log) channel operable. Specifically, the

licensee did not take action to suspend operations involving diluted water additions to the volume control tank and temperature increases with a positive moderator temperature coefficient (MTC) without the required number of operable log channels. As a result, the licensee did not comply with Action 4 of TS LCO 3.3.1 because they did not suspend all operations involving positive reactivity changes with the exception of minimum reactivity additions due to temperature fluctuations or operations, which are necessary to maintain fluid inventory. The licensee entered this issue into their corrective action program as CR-WF3-2013-2166 and CR-WF3-2013-3182. The immediate corrective actions taken to restore compliance included the discontinued use of water additions to the volume control tank and the increase of RCS temperatures with a positive MTC until the licensee's personnel returned an additional log channel to service.

The failure to comply with TS LCO 3.3.1, Action 4, was a performance deficiency. The performance deficiency was more than minor because it was associated with the configuration control attribute of the Barrier Integrity cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the LCO for the log power channels ensures that adequate information is available to verify core reactivity conditions while shutdown to minimize the probability of the occurrence of postulated events. The inspectors used Checklist 4 contained in Attachment 1 of the NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists," to evaluate this finding. The inspectors determined that the finding did not meet the reactivity guidelines because the licensee did not comply with TS LCO 3.3.1, Action 4. The inspectors determined that the finding was of very low safety significance (Green) because it did not require a quantitative assessment and was not similar to any of the examples requiring a phase two or phase three analyses. The inspectors also determined that the licensee maintain the required shutdown margin to preclude inadvertent criticality in the shutdown condition. The inspectors concluded that the finding reflected current licensee performance and involved a cross-cutting aspect in the decision-making component of the human performance area in that the licensee did not make a safety-significant decision using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety was maintained. This included obtaining interdisciplinary input and reviews on safety-significant decisions [H.1.a] (Section 1R20).

B. Licensee-Identified Violations

None

PLANT STATUS

The Waterford Steam Electric Station, Unit 3, began the inspection period at approximately 100 percent power. On April 26, 2013, operators commenced a reactor shutdown to repair leaks associated with the safety injection tank 2B and a volume control tank sampling line, respectively. On April 30, operators began to raise power to 100 percent. On May 1, the unit reached 100 percent power and remained there 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 Summer Readiness for Offsite and Alternate-AC Power Systems

a. Inspection Scope

The inspectors performed a review of preparations for summer weather for selected systems, including conditions that could lead to loss-of-offsite power and conditions that could result from high temperatures. The inspectors reviewed the procedures affecting these areas and the communications protocols between the transmission system operator and the plant to verify that the appropriate information was being exchanged when issues arose that could affect the offsite power system. Examples of aspects considered in the inspectors' review included:

- The coordination between the transmission system operator and the plant's operations personnel during off-normal or emergency events
- The explanations for the events
- The estimates of when the offsite power system would be returned to a normal state
- The notifications from the transmission system operator to the plant when the offsite power system was returned to normal

During the inspection, the inspectors focused on plant-specific design features and the procedures used by plant personnel to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the UFSAR and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. Specific documents reviewed during this inspection are listed in the attachment. The inspectors also reviewed corrective action program items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems:

- 4160 volt alternating current electrical distribution
- 6900 volt alternating current electrical distribution

These activities constitute completion of one sample to evaluate the readiness of offsite and alternate-ac power for summer weather, as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

.2 Readiness to Cope with External Flooding

a. Inspection Scope

On January 16 and May 2, 2013, the inspectors reviewed external flooding for the reactor auxiliary building roof and cooling tower areas, respectively. The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flooding in these areas. The evaluation included a review to check for deviations from the descriptions provided in the UFSAR for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed an inspection of the protected area to identify any modification to the site that would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two external flooding samples, as defined in Inspection Procedure 71111.01-05.

b. Findings

Introduction. The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, because the licensee did not provide design control measures for verifying or checking the adequacy of the features designed to withstand the effects of a probable maximum precipitation (PMP) flooding event on the reactor auxiliary building (RAB) roof areas. Specifically, the licensee did not provide an analysis to demonstrate that adequate flood protection existed from the effects of a PMP flooding event on safety-related components and electrical equipment located on the roof of the RAB in the main steam isolation valve (MSIV) wing areas.

Description. During a walkdown of the reactor auxiliary building roof as a part of the independent assessment for the NRC Temporary Instruction 2515/187 – “Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns,” the inspectors identified safety-related components, electrical conduit, and power cables that were

vulnerable to the local PMP flooding event. The inspectors noted that the expected ponding levels from the event would challenge safety-related equipment such as the emergency feedwater flow control and isolation valves and cables, main steam isolation valves and cables, atmospheric dump valves, and back-up nitrogen accumulator components. The inspectors questioned whether the licensee performed an analysis to demonstrate that these safety-related components and electrical equipment located on the roof in the MSIV wing areas were protected against the effects of a PMP as described in Section C.3, Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants." At the time, the licensee could not identity any analysis that addressed the estimated depth of ponding from a local PMP on the RAB roof areas and how that ponding could affect safety-related equipment although the licensee committed to comply with Regulatory Guide 1.102, Revision 1. The inspectors documented this issue of concern as an unresolved item (URI 05000382/2012005-03) to determine if a performance deficiency existed or if the issue of concern constituted a violation. The licensee initiated a condition report and performed a search of their document control system for an analysis that addressed the requirements of their current licensing basis and regulatory guide commitments.

The inspectors reviewed the condition report and the licensee's current licensing basis documents in order to resolve the unresolved item. A review of the Waterford Steam Electric Station, Unit 3 Final Safety Analysis Report (FSAR) as updated identified that the licensee considered the effects of roof loading on several roof areas as described in section 2.4.2.3.3. However, the inspectors noted that the licensee could not locate any analysis that supported the total drainage capacity of the scuppers and roof drains installed on the RAB roof. In addition, the licensee did not have an analysis that addressed the estimated depth of ponding from a local PMP on the RAB roof areas and how that ponding could affect safety-related equipment. The inspectors determined that a performance deficiency existed in design control from the unresolved item because the licensee did not provide design control measures for verifying or checking the adequacy of the features designed to withstand the effects of a PMP flooding event on the RAB roof areas. The inspectors concluded that the licensee missed opportunities to identify this issue during the Hurricane Isaac event and the Fukushima Flooding Walkdowns. The immediate corrective actions taken to restore compliance included the performance of a preliminary analysis to demonstrate that the installed scuppers and roof drains have margin to protect against a local PMP flooding event and that the ponding depth would have little or no affect on the safety-related equipment and cables located in the MSIV wing areas. The licensee entered this issue into their corrective action program as CR-WF3-2012-7520.

Analysis. The failure to provide design control measures for verifying or checking the adequacy of the features designed to withstand the effects of a local PMP on RAB roof areas was a performance deficiency. The inspectors determined that it was reasonable for the licensee to be able to foresee and prevent the occurrence of this deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating System cornerstone and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the safety-related equipment located on the RAB roof areas are required to safely shutdown and maintain the reactor in a cold shutdown condition following accidents and anticipated operational occurrences. The inspectors used the NRC Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings" and Appendix A, "The Significance

Determination Process (SDP) for Findings At-Power," to evaluate this issue. The inspectors determined that the finding was of very low safety significance (Green) because it did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding, or severe weather initiating event. The inspectors concluded that the finding reflected current licensee performance and involved a cross-cutting aspect in the corrective action program component of the problem identification and resolution area in that the licensee did not identify potential flooding issues completely, accurately, and in a timely manner commensurate with their safety significance [P.1(a)].

Enforcement. Title 10 of CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, as of December 12, 2012, the licensee did not provide adequate design control measures for verifying or checking the adequacy of the features designed to withstand the effects of a PMP flooding event on the RAB roof areas. Specifically, the licensee did not provide an analysis to demonstrate that adequate flood protection existed from the effects of a PMP flooding event on safety-related components and electrical equipment located on the roof of the RAB in the MSIV wing areas. As a result, the licensee had to perform an analysis to determine if expected ponding levels from a PMP flooding event would challenge safety-related components and electrical equipment. The licensee entered this condition into their corrective action program as CR-WF3-2012-7520. The immediate corrective actions taken to restore compliance included the performance of a preliminary analysis to show that the installed scuppers and roof drains have margin to protect against a local PMP flooding event and that the ponding depth would have little or no affect on the safety-related equipment and cables located in the MSIV wing areas. Because this violation of Appendix B, Criterion III was of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000382/2013003-01, "Failure to provide design control measures to withstand the effects of flooding on the reactor auxiliary building roof."

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- On April 10, 2013, emergency diesel generator train A during a planned surveillance run of the emergency diesel generator train B
- On May 2, 2013, main steam isolation train B during planned maintenance on the main steam isolation valve number 1
- On June 13, 2013, essential chiller train A during emergent maintenance on essential chiller B

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also inspected accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three partial system walkdown samples as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

On June 18, 2013, the inspectors performed a complete system alignment inspection of the emergency diesel generator B room ventilation system to verify the functional capability of the system. The inspectors selected this system based on risk-informed insights from site-specific risk studies together with other factors, such as engineering analysis and judgment, operating experience, performance history, current plant mode, and/or previous walkdowns. The inspectors reviewed plant procedures, including abnormal and emergency, drawings, USAR and vendor manuals to determine the correct lineup and visually inspected the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. The inspectors reviewed a sample of past and outstanding work orders to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment-alignment problems were being identified and appropriately resolved. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one complete system walkdown sample as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- On April 4, 2013, reactor auxiliary building, fire area 3, heating ventilation & air conditioning (HVAC) equipment room
- On May 22, 2013, reactor auxiliary building, fire area 13, battery room 3A
- On June 19, 2013, reactor auxiliary building, fire area 15, emergency diesel generator room (EDG) (3B)
- On June 19, 2013, reactor auxiliary building, fire area 16, emergency diesel generator (EDG) room 3A

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four quarterly fire-protection inspection samples, as defined in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On May 9, 2013, the inspectors observed a fire brigade activation of an actual fire in the water treatment building, fire area NS-WTB. The observation evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies; openly discussed them in a self-critical manner, and took appropriate corrective actions. Specific attributes evaluated were (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient firefighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; and (8) utilization of preplanned strategies.

These activities constitute completion of one annual fire-protection inspection sample, as defined in Inspection Procedure 71111.05-05.

b. Findings

No findings were identified.

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

.1 Quarterly Review of Licensed Operator Requalification Program

a. Inspection Scope

On June 13, 2013, the inspectors observed a crew of licensed operators in the plant's simulator during training. The inspectors assessed the following areas:

- Licensed operator performance
- The modeling and performance of the control room simulator
- The quality of post-scenario critiques
- Follow-up actions taken by the licensee for identified discrepancies

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

b. Findings

No findings were identified.

.2 Quarterly Observation of Licensed Operator Performance

a. Inspection Scope

On April 30, 2013, the inspectors observed the performance of on-shift licensed operators in the plant's main control room during start-up activities from a planned force outage. At the time of the observations, the plant was in a period of heightened awareness due to start-up activities related to reactivity control. The inspectors observed the operators' performance of the following activities:

- Start-up activities
- Reactivity control

In addition, the inspectors assessed the operators' adherence to plant procedures, including the conduct of operations procedure and other reactivity management policies and procedures.

These activities constitute 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)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant systems:

- On April 24, 2013, emergency diesel generator air compressors
- On May 8, 2013, condensate makeup valves CMU-908 and CMU-909

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance

- Trending key parameters for condition monitoring

- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or -(a)(2)

- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two maintenance effectiveness samples, as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed licensee personnel's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- On April 2, 2013, scheduled maintenance on train A essential switchgear air handling unit (AHU-2)

- On April 16, 2013,scheduled maintenance on train B emergency diesel generator with a tornado watch in effect

- On May 10, 2013, emergent maintenance on train B control room envelope with a tornado warning in effect

- On June 18, 2013, emergent maintenance on train B emergency diesel generator with the AB component cooling pump being out of service.

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance

work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four maintenance risk assessments and emergent work control inspection samples, as defined in Inspection Procedure 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following assessments:

- On April 3, 2013, essential switchgear air handling unit fan train A
- On April 4, 2013, high pressure safety injection train AB full flow recirculation with pressurization of train A during surveillance testing
- On April 11, 2013, failure of both trains of fuel building air handling units
- On May 2, 2013, hot leg injection valve 512B leak-by
- On May 10, 2013, control room envelop pressure test being unsatisfactory
- On June 18, 2013, emergency diesel generator train B failed to achieve required voltage and frequency within 10 seconds

The inspectors selected these operability and functionality assessments based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure technical specification operability was properly justified and to verify the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and UFSAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six operability evaluations inspection samples, as defined in Inspection Procedure 71111.15-05.

b. Findings

No findings were identified.

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

.1 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed ten evaluations to determine whether the changes to the facility or procedures, as described in the UFSAR, had been reviewed and documented in accordance with 10 CFR 50.59 requirements. The inspectors verified that when changes, tests, or experiments were made, evaluations were performed in accordance with 10 CFR 50.59 and licensee personnel had appropriately concluded that the change, test, or experiment could be accomplished without obtaining a license amendment. The inspectors also verified that safety issues related to the changes, tests, or experiments were resolved. The team compared the safety evaluations and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR Part 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluations.

The inspectors reviewed eighteen samples of changes, tests, and experiments that licensee personnel determined did not require evaluations and verified that the licensee personnel's conclusions were correct and consistent with 10 CFR 50.59.

The inspectors also verified that calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications, and plant drawings used to support the changes were accurate after the changes had been made. Documents reviewed are listed in the attachment.

These activities constitute completion of ten samples of evaluations and eighteen samples of changes, tests, and experiments that were screened out by licensee personnel as defined in Inspection Procedure 71111.17-04.

b. Findings

No findings were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The inspectors verified that calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications, and plant drawings used to support the modifications were accurate after the modifications had been made. The inspectors verified that modifications were consistent with the plant's licensing and design basis. The inspectors confirmed that revised calculations and analyses

demonstrated that the modifications did not adversely impact plant safety. Additionally, inspectors interviewed design and system engineers to assess the adequacy of the modifications.

These activities constitute completion of eighteen samples of permanent plant modifications as defined in Inspection Procedure 71111.17 04, and specific documents reviewed during this inspection are listed in the attachment.

.2.1 Add Vibration Dampener to Class 5 Pipe

The inspectors reviewed Engineering Change 42373, implemented to add a vibration dampener to feedwater pipe between two valves. The addition of the clamp was intended to reduce the vibrations seen on the pipe and valves and installed under Nuclear Change 42373. The modification was completed, additional vibration readings were taken, and subsequently the clamp was removed when the vibrations were not reduced. The inspectors reviewed the supporting drawings and design basis documents to verify that the associated programs were updated.

.2.2 Add Vibration Dampener to Class 2 Pipe

The inspectors reviewed Engineering Change 42374, implemented to add a vibration dampener to feedwater pipe between two valves inside containment. The addition of the clamp was intended to reduce the vibrations seen on the pipe and valves and installed under Nuclear Change 42374. The modification was completed, additional vibration readings taken, and subsequently the clamp was removed when the vibrations were not reduced. The inspectors reviewed the supporting drawings and interviewed licensee personnel to verify that the associated programs and design basis documents were updated.

.2.3 Miscellaneous Structural Modifications Inside Containment for Steam Generator and Reactor Vessel Head Replacement

The inspectors reviewed Engineering Change 8428, implemented to relocate several jib cranes mounted to structural steel beams and columns inside containment. The modification removed items that would interfere with removal of the old steam generators and installation of the new steam generator and reactor vessel head. Additionally, the engineering change included justification for allowing some scaffolding and cabling to remain inside containment between refueling cycles due to the postponement of the steam generator and reactor vessel head replacement. The inspectors reviewed the supporting drawings and interviewed licensee personnel to verify that the associated programs and design basis documents were updated.

.2.4 Modify Upper Guide Structure for ICI Thimble

The inspectors reviewed EC 10454, implemented to modify the upper guide structure lifting rig that is used during refueling outages to assist in removal of the control rod drive mechanisms. The modification included shortening a pin and relocating the handrails in order to eliminate interference with the upper plate. The inspectors reviewed the supporting drawings and interviewed licensee personnel to verify that the associated programs and design basis documents were updated.

.2.5 Move Trip Switch Plate for Cask Crane

The inspectors reviewed EC 30731, implemented to relocate the trip switch plate associated with the fuel handling cask crane. The trip switch plate was relocated so that the hook was allowed to reach the dry cask storage and handling areas, but not extend over the spent fuel pool. The trip switch plate is not considered integral to the crane, but acts to disconnect power should the crane be positioned past its referenced safety point. The inspectors questioned if the plate should be seismically mounted, but determined seismic qualification was not required because it cannot fail other safety related components or systems and an internal vibration sensor can trip the crane during a seismic event. The inspectors also reviewed the affected documentation to ensure that the licensee updated the documents as applicable.

.2.6 Modify Operating Procedures for New Steam Generator Tubing Limits

The inspectors reviewed EC 24060, implemented to support new pressure and temperature limits on normal startup and cooldown implemented to comply with Regulatory Guide 1.121 and the technical specification steam generator tube plugging limits. This modification included adding a new pressure/temperature limit graph and cautionary language to five operating procedures. The inspectors verified that the operating procedures were modified to include the new pressure and temperature limits, and interviewed control room operators regarding the new limits.

.2.7 Core Power Calculator Constant Update

The inspectors reviewed EC 7436, implemented to support new pressure and temperature limits on normal startup and cooldown implemented to comply with Regulatory Guide 1.121 and the technical specification steam generator tube plugging limits. This modification included modification of several core power calculator addressable constants to reflect the new fuel cycle limits. The inspectors verified that the revised documents included the new limits and the safety analysis provided by Westinghouse within NRC codes and methods.

.2.8 Reactor Coolant System Pressure/Temperature Limits

The inspectors reviewed EC 41604, implemented to revise reactor coolant pressure and temperature limits on normal startup and cooldown implemented to comply with Regulatory Guide 1.121 and the technical specification steam generator tube plugging limits. This modification included changing the operating limits associated with the operation of the reactor coolant pumps. The inspectors verified that the revised documents included the new pressure and temperature limits in order to ensure net positive suction head on the pumps.

.2.9 Reactor Auxiliary Building Room Area Temperature Monitoring Instrument Ranges

The inspectors reviewed EC 5000084913, which revised the calibrated range and temperature setpoints of the instrumentation loops for multiple reactor auxiliary building rooms that housed safety related equipment. The inspectors verified that the calibration procedures were updated and the new setpoints were conservative with respect to the design basis temperature limits.

.2.10 TS Bases 3/4.3.1-3/4.3.2 4-kV Safety Bus Relay Clarification

The inspectors reviewed EC 12084, which revised the Technical Specifications Bases Section 3/4.3.1 and 3/4.3.2 for reactor protective and engineered safety feature actuation systems instrumentation. The inspectors verified that the change to the technical specification bases did not necessitate a license amendment.

.2.11 Backup Instrument Air Supply for SI-129A(B) to Throttle

The inspectors reviewed EC 30976, which installed a safety-related backup in air supply accumulator to safety injection valve SI-129 A(B). The inspectors verified that the new pipe and reservoir met all applicable construction code requirements. Additionally, the inspectors reviewed the abnormal and emergency operating procedures to verify that proper manipulation of the backup instrument air would occur during accident conditions.

.2.12 Changes to EC-14765 for CALC MPR-2390, Shutdown Cooling Gas Intrusion Analysis, Revision 3

The inspectors reviewed EC 25944, which revised calculation ECM03-003, “Analysis of Waterford Station, Unit 3 SDC System Gas Accumulation,” by updating the valve flow coefficient for bypass valve SI-4052 A(B) and the addition of an orifice upstream of the bypass valve. The inspectors verified that the licensee’s revised gas accumulation evaluation continued to conclude low pressure safety injection system would still meet its design function.

.2.13 Revise Safety Injection Sump Strainer Design Basis to Address the Single Failure of a Low Pressure Safety Injection Pump to Automatically Stop on Recirculation Actuation Signal

The inspectors reviewed EC 26496, implemented to revise emergency operating procedure contingency actions to respond to the single failure of a low pressure safety injection pump to automatically stop when a recirculation actuation signal is received. The inspectors verified that emergency operation procedures were updated to include the manual action to stop a low pressure safety injection pump.

.2.14 Perform the Chilled Water Design Change Described in CR-WF3-2006-00440

The inspectors reviewed EC 2188, which was prepared to analyze the reduction in design chilled water flows for the AH-3A(B), AH-12A(B), AH-21, AH-26A(B) and AH-30A(B) safety related room coolers. The inspectors verified that the reduced chill water flow did not inhibit the chillers from performing their design basis functions.

.2.15 Fuel Pool Pumps A&B Water Level Switch Bypass, Rev. 1

The inspectors reviewed EC 18232, which implemented a temporary modification to install a temporary control switch in the control circuit for the spent fuel pool pumps A and B to bypass the spent fuel pool low-low level pump trip during RF 16 while SUPS AB bus was de-energized.

.2.16 Installation of Conduit in Containment to Facilitate RCP Seal Injection

The inspectors reviewed EC 12592, which permanently installed 4 - 2" conduits leading from the reactor containment building +21 feet elevation through the biological shield wall and terminating inside each reactor coolant pump shroud at approximate elevation of reactor containment building +18 feet elevation. This installation provides a means for performing a remote visual inspection of the reactor coolant pump shroud area during power operation.

.2.17 Install Replacement Timer Relays in Auxiliary Panels 1C and 2C

The inspectors reviewed EC 17647, which replaced Agastat E7024 relays with Allen Bradley 700RTC relays, and if required, a multiplying Allen Bradley 700-DC-P relay. These Agastat relays were being replaced due to aging and reliability concerns.

.2.18 Clarify Safety Function and Leakage Criteria for CS-117A(B)

The inspectors reviewed EC 31375, which replaced an existing 15-inch diameter hand wheel on containment spray valve CS-117A(B) with 60-inch diameter hand wheel, with torque limiter set at 237 ft-lbs. to provide leak tight closure of containment spray valve CS-117A(B) during shutdown cooling entry conditions. The inspectors reviewed the implementing work orders and post maintenance test results to ensure the valve provided a leak tight closure during shutdown cooling entry conditions.

b. Findings

Failure to Update Fuel Handling Accident Analysis in the Updated Final Safety Analysis Report

Introduction. The inspectors identified a Severity Level IV non-cited violation for the licensee's failure to update the final (updated) safety analysis report in accordance with 10 CFR 50.71(e). Specifically, the licensee failed to update the Waterford Steam Electric Station, Unit 3, UFSAR, Section 15.7.3.4 "Design Basis Fuel Handling Accidents," to appropriately reflect the fuel drop accident analysis methodology and impact stresses of the fuel bundles associated with Next Generation Fuel.

Description. On April 15, 2013, while reviewing the Waterford Steam Electric Station, Unit 3, UFSAR, Section 15.7.3.4 "Design Basis Fuel Handling Accidents," the inspectors noted that calculations used for determining the impact stress were dependent on the weight of the fuel bundle. Additionally, the inspectors noted no reference to a modification of that weight, when the licensee began incorporating Next Generation Fuel, which has a slightly lower fuel size and weight, in June 2008 during Operating Cycle 16.

When incorporating the use of Next Generation Fuel, the licensee consulted with Westinghouse to revise the analysis of record to reflect the revised impact stresses in the fuel rod cladding during a fuel bundle drop accident. Westinghouse verified the new figures were within the allowable tolerances of the analysis of record which did not result in fuel damage. Westinghouse recommended that several sections of the UFSAR were in need of updating to reflect the new fuel specifications; however, the section for fuel bundle drop accident analysis was not updated to reflect the introduction of Next Generation Fuel.

Additionally, the inspectors discovered that the current methodology used by Westinghouse to evaluate a fuel bundle drop accident was slightly different from the methodology described in the latest update to the Waterford Steam Electric Station, Unit 3, UFSAR, submitted to the NRC on May 9, 2012. The current methodology used by Westinghouse was reviewed and approved by the NRC in NUREG-0787, "Safety Evaluation Report related to the operation of Waterford Steam Electric Station, Unit No. 3," dated July 1981; however, the licensee failed to update the UFSAR to reflect the change to the fuel bundle drop analysis.

The licensee entered the condition into their corrective action program as Condition Report CR-WF3-2013-01931. The updates to the UFSAR have been drafted and submitted for inclusion in the next scheduled update.

Analysis. The failure to update the methodology, the data input to the calculation, and the resulting limits for the fuel bundle drop accident analysis in Section 15.7.3.4 of the UFSAR in accordance with 10 CFR 50.71(e) is a performance deficiency. This performance deficiency was evaluated using traditional enforcement because it has the potential to impact the NRC's ability to perform its regulatory function. The inspectors used the NRC Enforcement Policy to evaluate the significance of this violation. Consistent with Section 6.1.d.3 of the NRC Enforcement Policy, the inspectors determined that the performance deficiency is a Severity Level IV non-cited violation. This non-cited violation has no cross-cutting aspect because there was no finding associated with this traditional enforcement violations.

Enforcement. The inspectors identified a Severity Level IV non-cited violation of Title 10 CFR Part 50, paragraph 50.71(e), which states, in part, that each person licensed to operate a nuclear power reactor shall update, periodically, the FSAR originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the commission by the applicant or licensee or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original FSAR, or as appropriate, the last update to the FSAR under this section. The submittal shall include the effects of all changes made in the facility or procedures as described in the FSAR; all safety analyses and evaluations performed by the applicant or licensee either in support of approved license amendments or in support of conclusions that changes did not require a license amendment in accordance with §50.59(c)(2) or, in the case of a license that references a certified design, in accordance with § 52.98(c) of this chapter; and all analyses of new safety issues performed by or on behalf of the applicant or licensee at Commission request. The updated information shall be appropriately located within the update to the FSAR. Contrary to the above, from July 1981 to April 18, 2013, the licensee failed to update the methodology, the data input, and the resulting limits for the fuel bundle drop accident analysis in the Waterford, Unit 3, Updated Final Safety Analysis Report, Section 15.7.3.4, "Design Basis Fuel Handling Accidents." Because this is Severity Level IV violation, and was entered into the licensee's corrective action program as Condition Report CR-WF3-2013-0193, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 05000382/2013003-02, "Failure to Update Fuel Handling Accident Analysis in the Updated Final Safety Analysis Report."

1R18 Plant Modifications (71111.18)

Temporary Modifications

a. Inspection Scope

On April 26, 2013, to verify that the safety functions of important safety systems were not degraded, the inspectors reviewed the temporary modification EC-43821, which provided a regulated nitrogen supply to safety injection tank 2B.

The inspectors reviewed the temporary modification and the associated safety-evaluation screening against the system design bases documentation, including the UFSAR and the technical specifications, and verified that the modification did not adversely affect the system operability/availability. The inspectors also verified that the installation and restoration were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modification was identified on control room drawings, appropriate tags were placed on the affected equipment, and licensee personnel evaluated the combined effects on mitigating systems and the integrity of radiological barriers.

These activities constitute completion of one sample for temporary plant modifications, as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following post-maintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- On April 10, 2013, testing of the emergency diesel generator, train A, following planed maintenance
- On April 25, 2013, testing of the emergency diesel generator, train B, following planed maintenance
- On April 29, 2013, testing following emergent repairs on nitrogen supply valve NG-162B to safety injection tank 2B
- On May 2, 2013, testing following emergent repairs on volume control tank outlet sample isolation valve CVC-180
- On May 26, 2013, testing following replacement of the hub on the train B emergency diesel generator exhaust fan

- On June 7, 2013, testing following maintenance on controlled ventilation area system trains A and B

- On June 20, 2013, emergency diesel generator B following maintenance on manual air start system.

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated these activities for the following (as applicable):

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed

- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of seven post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19-05.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

Other Outage Activities

a. Inspection Scope

On April 26, 2013, Waterford Steam Electric Station, Unit 3, commenced a reactor shutdown to repair leaks associated with the safety injection tank 2B and a volume control tank sampling line, respectively. During this outage, the inspectors observed portions of the startup and heat-up processes, monitored licensee controls over the outage activities listed below.

- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.

- Controls over activities that could affect reactivity.

- Startup and ascension to full power operation, tracking of startup prerequisites, and reactor physics testing.

- Licensee identification and resolution of problems related to the outage activities.

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one outage inspection sample, as defined in Inspection Procedure 71111.20-05.

b. Findings

Introduction. The inspectors identified a Green non-cited violation of Waterford Steam Electric Station, Unit 3, Technical Specification (TS) Limiting Condition of Operation (LCO) 3.3.1 because the licensee did not take action to suspend operations that involved reactivity changes to accomplish startup activities with only one excore nuclear instrumentation (ENI) logarithmic (log) channel operable. Specifically, the licensee did not take action to suspend operations involving water additions to the volume control tank and temperature increases with a positive moderator temperature coefficient (MTC) without the required number of operable log channels.

Description. On April 30, 2013, while the plant was in hot shutdown (Mode 4), an inspector observed operation personnel raise temperature (heat-up) and add primary makeup water to the volume control tank (VCT) as a part of startup activities for the unit. At the time, the inspector noted that the ENI logarithmic power channels failed below the minimum required TS LCO for operable channels (A, B, C, and D). The TS LCO 3.3.1 requires, in part, that a minimum of two of the four logarithmic power level-high channels remain operable during shutdown with the protective system trip breakers in the open position for Modes 3, 4, and 5. If only one log channel remain operable than Action 4 of TS LCO 3.3.1 applies. TS LCO 3.3.1, Action 4, states, in part, that with the number of channels operable one less than required by the minimum channels operable requirement, suspend all operations involving positive reactivity changes. Action 4 has a note associated with it that states that limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated Shutdown Margin (SDM). Prior to the heat-up and diluting of the reactor coolant system (RCS), the licensee had just declared ENI log power channel (A) inoperable due to it acting erratically with two others channels (C) and (D) already inoperable because of previous failures. The inspectors questioned whether the licensee met the intent of TS LCO 3.3.1, Action 4 since the heat-up and the water addition to the RCS did not seem necessary to control temperature and maintain RCS inventory, respectively. Based on the question, the licensee discontinued the water additions and halted the heat-up of the RCS. The inspectors did note that the licensee performed an SDM verification seven hours prior to the primary makeup water addition to the VCT. The licensee entered this issue of concern into their corrective action program as CR-WF3-2013-2166.

As a follow-up to this issue, the inspectors reviewed operator logs and procedures, compared the TS requirement and bases documents to the NRC safety evaluation, and interviewed the licensee's operations, reactor engineering, and licensing personnel. The inspectors also reviewed the technical specification task force (TSTF) 286, revision 2, to obtain insights on discussions for limited positive reactivity additions while in shutdown modes. During the review of the operator logs, the inspectors identified another instance

while the plant was in cold shutdown (Mode 5) from April 28 through April 29, 2013, that the ENI log channels failed below the minimum required number for operable log channels for the particular mode. The inspectors noted that during that period, the licensee conducted heat-ups of the RCS with a positive moderator temperature coefficient and blended make-ups (of water and boric acid) to the VCT in order to accomplish start-up activities. The review of the NRC safety evaluation related to the licensee's TS change provided information on the proposed changes that conform closely to TSTF-286, revision 2. Section 2.0, Background, of the NRC safety evaluation states, in part, that TSTF-286 revises most of the actions to allow minimum reactivity additions due to temperature fluctuations or operations, which are necessary to maintain fluid inventory within the required shutdown margin, or refueling boron concentration, as applicable. Based on the review of the TSTF and NRC safety evaluation, the inspectors determined that the licensee did not meet the intent of TS LCO 3.3.1, Action 4, because raising RCS temperature with a positive MTC and diluting the RCS to accomplish startup activities did not constitute actions taken for temperature fluctuations or operations that were necessary to maintain fluid inventory.

In addition, the inspectors noted that the TS bases did not include information that provided this background information from the TSTF or the NRC safety evaluation. In interviews with the licensee's licensing personnel, the inspectors noted that the licensee inherently omitted this information from the TS bases document. The inspectors asked operations personnel if the Shift obtained interdisciplinary inputs and reviews from reactor engineering and licensing since the TS bases did not exist for this TS LCO Action prior to performing the above tasks. Operations personnel replied that the operations shift discussed TS 3.3.1 Action 4 internally prior to the water additions because they felt that it would not cause a significant RCS boron concentration change since they accounted for it in the shutdown margin calculation performed earlier in the day. The inspectors concluded that the finding reflected current licensee performance and involved a cross-cutting aspect in the decision-making component of the human performance area in that the licensee did not make a safety-significant decision using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety was maintained. This included obtaining interdisciplinary input and reviews on safety-significant decisions. The licensee entered this issue into their corrective action program as CR-WF3-2013-3004 and CR-WF3-2013-3182. The immediate corrective actions taken to restore compliance included the suspension of the water additions and heat-up of the RCS.

Analysis. The failure to comply with TS LCO 3.3.1, Action 4, was a performance deficiency. The inspectors determined that it was reasonable for the licensee to be able to foresee and prevent the occurrence of this deficiency. The performance deficiency was more than minor because it was associated with the configuration control attribute of the Barrier Integrity cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the LCO for the log power channels ensures that adequate information is available to verify core reactivity conditions while shutdown to minimize the probability of the occurrence of postulated events. The inspectors used Checklist 4 contained in Attachment 1 of the NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists," to evaluate this finding. The inspectors determined that the finding did not meet the reactivity guidelines because the licensee did not comply with TS LCO 3.3.1, Action 4. The inspectors determined that

the finding was of very low safety significance (Green) because it did not require a quantitative assessment and was not similar to any of the examples requiring a phase two or phase three analyses. The inspectors also determined that the licensee maintain the required shutdown margin to preclude inadvertent criticality in the shutdown condition.

The inspectors concluded that the finding reflected current licensee performance and involved a cross-cutting aspect in the decision-making component of the human performance area in that the licensee did not make a safety-significant decision using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety was maintained. This included obtaining interdisciplinary input and reviews on safety-significant decisions [H.1(a)].

Enforcement. Waterford Steam Electric Station, Unit 3, TS 3.3.1 requires, in part, that a minimum of two of the four logarithmic power level-high channels remain operable during shutdown with the protective system trip breakers in the open position for Modes 3, 4, and 5. If only one log channel remain operable than Action 4 of TS 3.3.1 applies. TS 3.3.1, Action 4, states, in part, that with the number of channels operable one less than required by the minimum channels operable requirement, suspend all operations involving positive reactivity changes. Action 4 has a note associated with it that states that limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated Shutdown Margin. Contrary to the above, between April 28 and April 30, 2013, the licensee did not comply with Action 4 of TS 3.3.1 during shutdown with the protective system trip breakers in the open position for Modes 4 and 5. Specifically, the licensee did not suspend all operations involving positive reactivity changes with the exception of minimum reactivity additions limited to plant cooldown or boron dilution due to temperature fluctuations or operations, which are necessary to maintain fluid inventory. The immediate corrective actions taken to restore compliance included the discontinued use of water additions to the volume control tank and the increase of RCS temperatures with a positive MTC until maintenance personnel returned an additional log channel to service. Because this violation was of very low safety significance and the licensee entered it into their corrective action program as CR-WF3-2013-2166, CR-WF3-2013-3004, and CR-WF3-2013-3182, this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000382/2013003-03; “Failure to comply with Action 4 of TS 3.3.1 during shutdown in Modes 4 and 5.”

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the UFSAR, procedure requirements, and technical specifications to ensure that the surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning

- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data
- Annunciators and alarms setpoints

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing.

- On April 1, 2013, high pressure safety injection, train AB, inservice test
- On April 10, 2013, emergency diesel generator, train A, surveillance test
- On April 24, 2013, reactor coolant system leakage detection, reactor coolant system surveillance test
- On May 8, 2013, category A seat leak for low pressure safety injection to reactor coolant loop auto vent containment isolation valve, containment isolation valve test
- On May 10, 2013, control room pressurization, surveillance test
- On June 13, 2013, component cooling water A/B pump, surveillance operability test

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on June 13, 2013, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the control room simulator, emergency operating facility, and technical support center where the drill was observed to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the attachment.

These activities constitute completion of one emergency preparedness drill observation sample, as defined in Inspection Procedure 71114.06-05.

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 Physical Protection

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the first Quarter 2013 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, “Performance Indicator Program.”

This review was performed as part of the inspectors’ normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings were identified.

.2 Reactor Coolant System Specific Activity (BI01)

a. Inspection Scope

On May 29, 2013, the inspectors sampled licensee submittals for the reactor coolant system specific activity performance indicator for the period from the third quarter 2012 through the first quarter 2013. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's reactor coolant system chemistry samples, technical specification requirements, issue reports, event reports, and NRC integrated inspection reports for the period of April 2012 through March 2013, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one reactor coolant system specific activity sample, as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.3 Reactor Coolant System Leakage (BI02)

a. Inspection Scope

On April 24, 2013, the inspectors sampled licensee submittals for the reactor coolant system leakage performance indicator for the period from the fourth quarter 2012 through the first quarter 2013. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator logs, reactor coolant system leakage tracking data, issue reports, event reports, and NRC integrated inspection reports for the period of April 2012 through March 2013, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one reactor coolant system leakage sample, as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors reviewed attributes that included the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions. Minor issues entered into the licensee's corrective action program because of the inspectors' observations are included in the attached list of documents reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. The inspectors accomplished this through review of the station's daily corrective action documents.

The inspectors performed these daily reviews as part of their daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2, above, licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of December 2012 through June 2013 although some examples expanded beyond those dates where the scope of the trend warranted.

The inspectors also included issues documented outside the normal corrective action program in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

These activities constitute completion of one single semi-annual trend inspection sample, as defined in Inspection Procedure 71152-05.

b. Findings

No findings were identified.

.4 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors recognized a corrective action item documenting actions associated with the local manual hand-wheel function of several safety related air operated valves in the event their safety related air supply accumulators were exhausted. Specifically, the inspectors reviewed the past operability reviews and reportability evaluations. The inspectors reviewed the appropriateness of the assigned significance, the scope and depth of the causal analysis, and the timeliness of resolution. The inspectors assessed whether the evaluation identified likely causes for the issue and identified appropriate corrective actions to address the identified causes.

These activities constitute completion of one in-depth problem identification and resolution sample, as defined in Inspection Procedure 71152-05.

b. Findings

Introduction. The inspectors identified a Severity Level IV non-cited violation of 10 CFR 50.73(a)(1) because the licensee did not submit a Licensee Event Report (LER) in a timely manner after the discovery of a reportable event. Specifically, the licensee failed to submit a required LER within 60 days after the discovery of a condition that affected

the manual hand-wheel operation of safety related air operated valves following a loss of their corresponding back-up nitrogen accumulators.

Description. On November 12, 2012, the licensee discovered that the manual hand-wheel override on the header B auxiliary component cooling essential chiller isolation valve (ACC-112B) would not change the position of the valve. The licensee initiated a condition report and performed an apparent cause evaluation, extent of condition review and a reportability evaluation. The condition report documented that the licensee did not perform testing on the local manual hydraulic control system for the ACC-112B valve and several other safety-related air operated valves (AOVs). As a part of the immediate corrective actions, the licensee performed as found testing on some of the AOVs identified in the extent of condition review. The inspectors noted that the manual hand-wheel operation for emergency feedwater valves (EFW-223B) and (EFW-229A) did not function as design and the licensee personnel declared the valves inoperable. The licensee entered these issues into their corrective action program as CR-WF3-2012-7566 and CR-WF3-2012-7570. The inspectors identified and documented a non-cited violation 05000382/2013002-03 related to the failure of the licensee to identify and perform testing to demonstrate the local manual operation of these safety-related valves.

As a part of this issue, the inspectors reviewed the past operability and reportability evaluations associated with the condition reports written. The inspectors noted a number of performance deficiencies related to the licensee's past operability and reportability conclusions. For example, the licensee concluded that since the critical component failure evaluation considered EFW-223B and EFW-229A available, then the valves did not require a past operability evaluation. Moreover, the licensee did not perform testing or evaluate past operability on some of the valves because of a modification to install a backup air supply to the nitrogen accumulators. The licensee concluded that this eliminated the need for manual hand-wheel operation and a past operability review. Another example related to the reportability reviews was when the licensee concluded that operators would not perform the local manual operation of the ACC-112B valve until 7 days post-accident and that "it is likely that personnel would have arranged to recharge their respective safety-related accumulator" and therefore considered the valve operable. The inspectors questioned several of the licensee conclusions, which caused the licensee to re-evaluate the reportability reviews. The results of the review determined that a number of reportable conditions existed per the requirements of 10 CFR Part 50.73. The inspectors determined that the licensee did not evaluate the loss of functional capability of the AOVs to provide an adequate conclusion on past operability of the components within three years prior to the time of discovery. The inspectors concluded that the licensee did not review for the applicable reportable condition as described in 10 CFR Part 73 and NUREG-1022, "Event Reporting Guidelines, 10 CFR 50.72 and 50.73." The licensee entered this issue into their corrective action program as CR-WF3-2013-2564. The immediate corrective actions taken to restore compliance included a new reportability review of the condition and the development of an LER.

Analysis. The failure to submit a required LER within 60 days after discovery of a condition that required a report was a violation of NRC requirements. The inspectors also determined that this violation was a performance deficiency. The inspectors determined that it was reasonable for the licensee to be able to foresee and prevent the occurrence of this deficiency. However, the inspectors determined that the performance deficiency was minor. The inspectors considered this issue to be within the traditional

enforcement process because it had the potential to impact the NRC's ability to perform its regulatory oversight function. The inspectors used the NRC Enforcement Policy to evaluate the significance of this violation. The inspectors determined that the violation was a Severity Level IV because it was similar to an example provided in Section 6.9 of the NRC Enforcement Policy (i.e., A licensee fails to make a report required by 10 CFR 50.73). The inspectors did not assign a cross-cutting aspect to this violation because there was no finding associated with this traditional enforcement violation

Enforcement. Title 10 of the CFR 50.73(a)(1) requires, in part, that licensees shall submit a LER for any event described in this paragraph within 60 days after discovery of the event. Contrary to the above, as of May 23, 2013, the licensee failed to submit an LER for an event described in 10 CFR 50.73 within 60 days after the discovery of the event. Specifically, the licensee failed to submit a required LER within 60 days after the discovery of a condition on November 12, 2012 that affected the manual hand-wheel operation of safety related air operated valves following a loss of their corresponding back-up nitrogen accumulators. The immediate corrective actions taken to restore compliance on May 23, 2013 included a re-evaluation of the event and an upcoming submittal of an LER. Because this violation was of very low safety significance and the licensee entered it into their corrective action program as CR-WF3-2013-2564, this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000382/2013003-04; "Failure to submit an LER after discovery that manual hand-wheels of AOVs were not functional."

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

(Closed) Licensee Event Report 05000382/2013-001-00 Main Feedwater Regulating Valve Closure Results in Automatic Reactor Trip

On January 21, 2013, the unit experienced an automatic reactor trip from 91 percent power due to a secondary transient that cause a low level in the number one steam generator. The secondary transient was due to an unexpected closure of the number 1 main feedwater regulating valve because of a vibration-induced failure of its instrument air supply line. As a part of the review of this event, the inspectors identified a non-cited violation 05000382/2013003-05, "Failure to provide adequate post modification testing instructions for vibration monitoring on the feedwater piping system." The inspectors documented this violation in Section 4OA5 of this report. This LER is closed.

4OA5 Other Activities

.1 (Discussed) Temporary Instructions 2515/182 - Review of the Industry Initiative to Control Degradation of Underground Piping and Tanks

a. Inspection Scope

Leakage from buried and underground pipes has resulted in ground water contamination incidents with associated heightened NRC and public interest. The industry issued a guidance document, Nuclear Energy Institute (NEI) 09-14, "Guideline for the Management of Buried Piping Integrity" (ADAMS Accession No. ML1030901420) to describe the goals and required actions (commitments made by the licensee) resulting from this underground piping and tank initiative. On December 31, 2010, NEI issued

Revision 1 to NEI 09-14, "Guidance for the Management of Underground Piping and Tank Integrity," (ADAMS Accession No. ML110700122), with an expanded scope of components which included underground piping that was not in direct contact with the soil and underground tanks. On November 17, 2011, the NRC issued TI-2515/182 "Review of the Industry Initiative to Control Degradation of Underground Piping and Tanks" to gather information related to the industry's implementation of this initiative. The inspectors reviewed the licensee's programs for buried pipe, underground piping and tanks in accordance with TI-2515/182 to determine if the program attributes and completion dates identified in Sections 3.3.A and 3.3.B of NEI 09-14 Revision 1 were contained in the licensee's program and implementing procedures. For the buried pipe and underground piping program attributes with completion dates that had passed, the inspectors reviewed records to determine if the attribute was in fact complete and to determine if the attribute was accomplished in a manner which reflected good or poor practices in program management.

b. Findings and Observations

Based upon the scope described above, Phase I was found to meet all applicable aspects of NEI 09-14, Revision 1, as set forth in Table 1 of TI-2515/182.

.2 (Closed) NRC Temporary Instruction 2515/187 - Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

The inspectors verified that the licensee's walkdown packages contained the elements as specified in NEI 12-07 Walkdown Guidance document.

The inspectors performed an independent walkdown of the reactor auxiliary building roof areas and verified that the licensee confirmed the following flood protection features:

- Visual inspection of the flood protection feature was performed if the flood protection feature was relevant. External visual inspection for indications of degradation that would prevent its credited function from being performed
- Reasonable simulation, if applicable to the site
- Critical SSC dimensions were measured
- Available physical margin, where applicable, was determined
- Flood protection feature functionality was determined using either visual observation or by review of other documents.

b. Findings

The inspectors identified an unresolved item (URI) related to a PMP flooding event that could potentially affect safety-related equipment located on the roof of the reactor auxiliary building in NRC inspection report 05000382/2012005-03. The inspectors resolved this item as a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, Design Control, and documented the results of the finding in Section 1R01 of this report.

.3 (Closed) URI 05000382/2012005-03 Reactor Auxiliary Building Roof Flood Protection Issue

The inspectors opened this unresolved item to determine if there was a performance deficiency associated with design control since the licensee did not have an analysis demonstrating adequate flooding protection for the RAB roof areas. The inspectors resolved this item as a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, Design Control, and documented the results in this report in Section 1R01. This URI is closed.

.4 Steam Generators Post Installation Verification and Testing

a. Inspection Scope

The inspectors reviewed post installation verification procedures for equipment performance testing required to confirm the design and to establish baseline measurements. Additionally, the inspectors reviewed procedures and engineering change documentation for how the licensee plans on conducting the post installation verification testing during power ascension.

b. Findings

Introduction. The inspectors reviewed a Green self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, because the licensee did not provide post modification testing instructions for activities affecting quality that were appropriate to the circumstances and that included appropriate acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee did not provide adequate post modification testing instructions for vibration monitoring of the feedwater piping system that included appropriate acceptance criteria and actions following the installation of the new replacement steam generators.

Description. On January 21, 2013, after the refueling outage (RFO-18) that installed two new Replacement Steam Generators (RSGs), the plant experienced an automatic reactor trip on low steam generator level. The plant tripped during the power ascension to 100 percent when an air line on the train A main feedwater regulating valve (FW-173A) failed. The licensee determined the direct cause of the air line failure was due to increased vibrations from the installation of the RSGs. The licensee initiated a condition report, performed repairs on the FW-173A air line, and revised the post modification testing instructions for the vibration monitoring of the feedwater piping system. Following the repairs on the FW-173A valve, the licensee restarted the plant and shortly after reaching 100 percent power on January 25, 2013, the licensee identified another leak on the same air line of the feedwater regulating valve. The licensee commenced a down power to 81 percent to reduce the vibrations reading on the feedwater piping in order to develop a more permanent solution for the air line leaks and vibration issues. The licensee included the repeat failure as a part of their root cause evaluation.

The inspectors performed a review of the event timeline, root cause evaluation, post modification testing instructions, and the revised vibration monitoring plan. The inspectors noted that the licensee developed a generic vibration monitoring program for the initial post modification testing instructions with little or no monitoring until the plant

reached 100 percent power. The objective of the initial instructions was to monitor vibrations of some of the plant systems during power ascension to 100 percent power and evaluate any significant changes that occurred due to the RSGs. However, during the initial startup from RF18, no piping vibrations were taken at intermediate power levels. The inspectors also noted that the initial instructions did not have any clear actions once the acceptance criteria were exceed for specific components and system piping. A review of the data after the reactor trip identified that a number of components and systems exceed the original vibration acceptance criterion of 0.5 inch per seconds with no corresponding evaluation. The inspectors determined that the licensee did not provide adequate post modification testing instructions for vibration monitoring of the feedwater piping system that included appropriate acceptance criteria and actions following the installation of the new RSGs. The licensee entered this issue into their corrective action program as CR-WF3-2013-0445. The immediate corrective actions taken to restore compliance included the implementation of a revised vibration-monitoring plan and the development of engineering changes to mitigate the vibration effects on the feedwater piping system.

In addition, the inspectors noted that the licensee had an opportunity to implement operating experience related to RSG vibration issues. The licensee had knowledge of previous RSG vibration issues from Arkansas Nuclear One (ANO), an Entergy South plant. The licensee performed an evaluation prior to installing the RSG using the ANO data but determined that a substantial rise in vibrations would have a low probability. The licensee did not use internally generated operating experience to support plant safety. The inspectors concluded that the finding reflected current licensee performance and involved a cross-cutting aspect in the operating experience component of the problem identification and resolution area in that the licensee did not implement operating experience through changes to station equipment to support plant safety.

Analysis. The failure to provide adequate post modification testing instructions for vibration monitoring of feedwater piping system following steam generator replacement was a performance deficiency. The inspectors determined that it was reasonable for the licensee to be able to foresee and prevent the occurrence of this deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations. The inspectors used the NRC Inspection Manual Chapter 0609, Attachment 4, “Initial Characterization of Findings” and Appendix A, “The Significance Determination Process for Findings At-Power,” to evaluate this issue. The inspectors determined that the finding was of very low safety significance (Green) because the transient initiator did not also contribute to the likelihood that mitigation equipment or functions would not be available. The inspectors concluded that the finding reflected current licensee performance and involved a cross-cutting aspect in the operating experience component of the problem identification and resolution area in that the licensee did not implement internal operating experience through changes to station equipment to support plant safety [P.2.b].

Enforcement. Title 10 of CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, as of December 12, 2012, the licensee did not provide post modification testing instructions for activities affecting quality that were appropriate to the circumstances and that included appropriate acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee did not provide adequate post modification testing instructions for vibration monitoring and mitigation of problems related to the feedwater piping system that included appropriate acceptance criteria following the installation of the new replacement steam generators. As a result, the plant experienced an automatic reactor trip and a subsequent down power because of vibration issues on the feedwater piping system. The licensee entered this issue into their corrective action program as CR-WF3-2013-0445. The immediate corrective actions taken to restore compliance included the implementation of a revised vibration-monitoring plan and the development of engineering changes to mitigate vibration effects on the feedwater piping system. Because this violation was of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000382/2013002-05, "Failure to provide post modification testing instructions for vibration monitoring on the feedwater piping system."

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 18, 2013, the inspectors presented the preliminary Triennial Permanent Plant Modifications & 50.59 inspection results to Mr. K. Nichols, Director of Engineering, and other members of the licensee's staff. The licensee acknowledged the results as presented. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

On July 11, 2013, the inspectors presented the inspection results to Mr. Carl Rich Jr., Director, Nuclear Safety Assurance, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

B. Lanka, Acting Director, Engineering
B. Lindsey, Manager, Operations
B. Pellegrin, Acting Manager Licensing
C. Fugate, Manager, Licensing
C. Rich, Jr., Director, Nuclear Safety Assurance
D. Boan, Supervisor, Radiation Protection
D. Frey, Manager Radiation Protection
D. Jacobs, Vice President, Operations
D. Reider, Supervisor, Quality Assurance
G. Pierce, Manager, Training
J. Briggs, Electrical Maintenance Superintendent
J. Frick, Security Manager
J. Russo, Supervisor, Design Engineering
K. Cook, General Manager, Plant Operations
K. Crissman, Manager, Maintenance
K. Nichols, Director, Engineering
M. Mason, Senior Licensing Specialist, Licensing
N. Lawless, Manager, Chemistry
P. Stanton, Supervisor, Design Engineering
R. Gilmore, Manager, Programs, Components and System Engineering
R. Porter, Manager, Design Engineering
S. Adams, Production Manager
S.W. Meiklejohn, I & C Maintenance Superintendent
W. Hardin, Senior Licensing Specialist, Licensing
W. McKinney, Manager, Corrective Action Programs

NRC Personnel

M. Davis, Senior Resident Inspector
R. Azua, Senior Project Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000382/2013003-01	NCV	Failure to provide design control measures to withstand the effects of flooding on the reactor auxiliary building roof (Section 1R01)
05000382/2013003-02	NCV	Failure to Update Fuel Handling Accident Analysis in the Updated Final Safety Analysis Report (Section 1R17)

05000382/2013003-03	NCV	Failure to comply with Action 4 of TS 3.3.1 during shutdown in Modes 4 and 5 (Section 1R20)
05000382/2013003-04	NCV	Failure to submit an LER after discovery that manual hand-wheels on AOVs were not functional (AOA2.4)
05000382/2013003-05	NCV	Failure to provide adequate post modification testing instructions for vibration monitoring on the feedwater piping system (Section AOA5.4)

Closed

05000382/2013-001-00	LER	Main Feedwater Regulating Valve Closure Results in Automatic Reactor Trip (Section 4OA3)
05000382/2012005-03	URI	Reactor auxiliary building roof flood protection issue of concern (Section 4OA5.3)
05000382/2515/187	TI	NRC Temporary Instruction 2515/187 - Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns (Section 4OA5.2)

Discussed

05000382/2515/182	TI	Temporary Instructions 2515/182 - Review of the Industry Initiative to Control Degradation of Underground Piping and Tanks (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION
OP-901-521	Severe Weather and Flooding	305
OP-901-314	Degraded Grid Operations	1
OP-006-001	Plant Distribution	311
OP-006-008	Transformer Operation	305
OP-006-009	Electrical Bus Outages	9
EC-C91-015	PEIR-20066 Evaluation of NRC Generic Letter 89-22, Revised PMP Criteria	0
EC-44973	EC to Document Evaluation of Ponding Level in MSIV Areas	0
EN-DC-170	Fukushima Near Term Task Force Recommendation 2.3 Flooding Walkdown Procedure	0

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION
WF3-CS-12-00004	Fukushima Flooding Walkdown Report	2

CONDITION REPORTS

CR-WF3-2012-5796 CR-WF3-2012-7520 CR-WF3-2013-1581

Section 1R04: Equipment Alignment

PROCEDURE

NUMBER	TITLE	REVISION
OP-009-002	Emergency Diesel Generator	320

CONDITION REPORTS

CR-WF3-2002-01414	CR-WF3-2003-01303	CR-WF3-2003-02441	CR-WF3-2004-01190
CR-WF3-2004-02003	CR-WF3-2006-04606	CR-WF3-2007-01713	CR-WF3-2007-03057
CR-WF3-2008-00753	CR-WF3-2012-01840	CR-WF3-2013-02530	

Section 1R05: Fire Protection

PROCEDURE

NUMBER	TITLE	REVISION
RAB 3-001	Pre-Fire Strategy Elevation +46.00' HVAC Equipment Room	007
UNT-007-060	Control of Loose Items	303
OP-009-004	Fire Protection	307
UNT-005-013	Fire Protection Program	11
FP-001-018	Pre-Fire Strategies, Development	301
FP-001-020	Fire Emergency/Fire Report	306
RAB 15-001	Waterford-3 S.E.S Pre-fire Strategy – Elev. +21.00' RAB (RCA) – Emergency Diesel Generator "3B"	8
RAB 16-001	Waterford-3 S.E.S Pre-fire Strategy – Elev. +21.00' RAB (RCA) – Emergency Diesel Generator "A"	11

CONDITION REPORTS

CR-WF3-2013-01700	CR-WF3-2013-01075	CR-WF3-2012-02879	CR-WF3-2013-01901
CR-WF3-2011-05424	CR-WF3-2011-06773	CR-WF3-2011-01477	CR-WF3-2012-03435

Section 1R11: Licensed Operator Requalification Program

PROCEDURES

NUMBER	TITLE	REVISION
EN-OP-115	Conduct of Operations	12
EN-RE-302	PWR Reactivity Maneuver	2
OP-004-019	Estimated Critical Configuration	12
OP-010-003	Plant Startup	330

Section 1R12: Maintenance Effectiveness

PROCEDURES

NUMBER	TITLE	REVISION
EN-DC-203	Maintenance Rule Program	1
EN-DC-204	Maintenance Rule Scope and Basis	2
EN-DC-153	Preventative Maintenance Component Classification	6
EN-DC-335	Preventative Maintenance Basis Document	4

CONDITION REPORTS

CR-WF3-2013-1185	CR-WF3-2013-2127	LO-WTWF3-2013-0098

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

PROCEDURES

NUMBER	TITLE	REVISION
EN-WM-104	On Line Risk Assessment	7
EN-OP-119	Protected Equipment Postings	5
OI-037-000	Operations' Risk Assessment Guideline	304
UNT-007-060	Control of Loose Items	303

CONDITION REPORT

CR-WF3-2013-01692

Section 1R15: Operability Evaluations

PROCEDURES

NUMBER	TITLE	REVISION
EN-OP-104	Operability Determination Process	6
OP-100-014	Technical Specifications and Technical Requirement Compliance	313

CONDITION REPORTS

CR-WF3-2013-1646	CR-WF3-2013-1452	CR-WF3-2013-1231	CR-WF3-2013-1258
CR-WF3-2013-2338	CR-WF3-2013-2946		

Section 1R17: Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

CALCULATIONS

NUMBER	TITLE	REVISION
2004-07780	Debris Generation due to LOCA Containment for Resolution of GSI-191	3
36569-19	NOG-1 Compliance Matrix P&H SUPERSAFE Single Failure Proof Crane for WSES-3 Waterford 3 Nuclear Power Station Fuel Handling Crane	1
5-D	HVAC Cooling Load for Cable Vault and Switchgear Area	2
5-T	Essential Chilled Water Cooling Loads & Coil Performance Determination	5
CN-OA-08-5	WSES-3 Loss of Shutdown Cooling from Mid-Loop	0
ECE91-058	Battery 3A-S “A” Train Calculation for Station Blackout	5
ECE91-059	Battery 3B-S “B” Train Calculation for Station Blackout	5
ECE91-061	Battery 3A-S Cell Sizing	5
ECE91-062	Battery 3B-S Cell Sizing	5
ECE91-193	Load Study for PDP-3A-DC-S, and 3A1-DC-S	4
ECE91-194	Load Study for PDP-3B-DC-S, and 3B1-DC-S	4
ECI01-009	Determination of HVAC Measurements Functional Safety Significance	1
ECM03-003	Shutdown Cooling Operation with Suction Piping Air	0
ECM10-003	Dry Cooling Tower B Air Flow Analysis for the SG/RVCH Replacement Project	0
ECM10-006	Dry Cooling Tower B Airflow Reduction Limits to Support Steam Generator Replacement	0
ECM12-002	Required Torque to Close and Seal CS-117A(B) and CS-111A(B)	0
ECM89-089	Allowable Instrument Air Accumulator Leak Rate	7

NUMBER	TITLE	REVISION
ECM95-008	Ultimate Sink Design Basis	3
ECM95-009	Ultimate Heat Sink Fan Requirements Under Various Ambient Conditions	2
ECS1O-002	Waterford 3 RCS Time-to-Boil Due to LOSDC at Various Levels and Temperature	0
SQ-IC-36	Seismic Qualification Review Team File for Auxiliary Panel	17
SQ-IC-80	Seismic Qualification Review Team File for Auxiliary Panel Electrical Components	07
SQ-IC-93	Seismic Qualification Review Team File for Allen Bradley 700DC-P Relays	00
SQ-IC-94	Seismic Qualification Review Team File for Allen Bradley 700RTC Relays	00
SQ-MN-209	ROTO Hammer Valve Extension Stems	0
WCAP-17623-P	Regulator Guide 1.121 Analysis and Structural Integrity Performance Criterion Application for the Waterford Unit 3 Model Delta 110 Replacement Steam Generators for a NSSS Power of 1869.6 MWt/SG	1

CONDITION REPORTS

CR-WF3-2010-05792	CR-WF-2012-12880	CR-WF3-2013-01907	CR-WF-2013-1381
CR-WF3-2013-01880	CR-WF3-2013-01763	CR-WF3-2013-01764	CR-WF3-2013-01797
CR-WF3-2013-01809	CR-WF3-2013-01817	CR-WF3-2013-01826	CR-WF3-2013-01831
CR-WF3-2013-01880	CR-WF3-2013-01904	CR-WF3-2013-01907	CR-WF3-2013-01834
CR-WF3-2013-01854	CR-WF3-2006-00440	CR-WF3-2011-05841	CR-WF3-2011-03350
CR-WF3-2013-01931	CR-WF3-2013-01941		

DRAWINGS

NUMBER	TITLE	REVISION
14EK-0144	Diesel Engine Generator Schematic Diagram	22
1564-1809	10#-300# Globe St/CHK VLV bolted Bonn, B, W, ENDG Bevel Gear Operator (5:1 Ratio)	7
1564-G246-S02, Sheet 2	Reactor Containment Building Section & Details	8
1564-G246-S02, Sheet 4	Reactor Containment Building Section & Details	8
2006, Sheet 1	Burns & Roe, Flow Diagram – Circulating, Screen Wash & Service Water Systems	N83
2011, Sheet 1	Burns & Roe, Flow Diagram Turbine Oil Purification & Transfer Sys & Diesel Oil Sys	N44
2038, Sheet1	Flow Diagram Reactor Building Floor & Roof Drain Systems	N54
2077	Burns & Roe, Flow Diagram – Diesel Gen. Bldg. Service Water, Starting Air, Fuel Oil, Sump System & Roof Drains	N77
3001	Main One Line Diagram	N22
3002	Auxiliary One Line Diagram MCC Z, SWGR BUS 1A, 1B, 1E, & Critical SWGR Bus 1F 1G	49
3203	Turbine Generator Building Mezzanine- Instrumentation Conduit & Tray Plan	N28

NUMBER	TITLE	REVISION
B288, Sheet 541	Cable and Conduit list Installation Notes Unique Seismic Supports	9
B288CCL	Cable and Conduit List (CCL)	0
B424, Sheet 1026	Control Wiring Diagram, Equipment Room Cooler AH-26(3A-SA)	12
B424, Sheet 1242	Control Wiring Diagram, SWGR AREA Air Handling Unit AH-30 (3A-SA)	15
B424, Sheet 1242	Control Wiring Diagram, SWGR AREA Air Handling Unit AH-30 (3A-SA)	13
B424, Sheet 2342	Control Wiring Diagram, Sequencer A, Sh. 2	13
B424, Sheet 2343	Control Wiring Diagram, Sequencer A, Sh. 3	12
B424, Sheet 2388	Control Wiring Diagram, 4.16KV Bus 3B3-S, Under Voltage Relays, Sheet 2	13
B430, Sheet V03A	Instrument Installation Details	2
B430, Sheet V03A	Instrument Installation	3
B434, Sheet 1133	Control Wiring Diagram, Containment Fan Cooler AH-1 (3S-SA)	19
B434, Sheet 1148	Control Wiring Diagram, Filtration Unit S-8 (3S-SA)	17
B434, Sheet 1165	Control Wiring Diagram, Control Room Air Handling Unit AH-12 (3A-SA)	23
B434, Sheet 2341	Control Wiring Diagram, Sequencer A Sh.1	13
B434, Sheet 631	Control Wiring Diagram, Fuel Pool Instrumentation	19
B434, Sheet 631	Control Wiring Diagram, Fuel Pool Instrumentation	19
B434, Sheet 635	Control Wiring Diagram, Fuel Pool Pump B	10
CNS-SW-35	SW A STR S-191 INSTR TUBING	N00
G1357	Fire Protection, Reactor Auxiliary Building Plan Elevation - 35.00	2
G167, Sheet 2	Safety Injection System	52
G167, Sheet 3	Safety Injection System	10
G5-262-743	Emergency Diesel Generator No. 1, Sh. 2	19
G853S08	HVAC – Air Flow Diagram, Sh-3	18
G853S22	HVAC Air Flow Diagram, Control Room	6
G853S23	HVAC Air Flow Diagram, Control Room	2
KSV-102-8	Fuel Oil Tank Outline	N02
KSV-72-24	Mechanical Butterfly Valve Overspeed Shutdown	N01
SKE 4699557-2	Assembly Drawing for Locking Open Device on DG Overspeed Valve	0
V-8.0-1003-65	Valve Extension Stem	B

50.59 SAFETY EVALUATIONS

2011-04	2011-03	2010-12	2008-02
2011-01	2008-03	2010-011	2012-01
2011-02	2009-02		

PROCEDURES

NUMBER	TITLE	REVISION
EN-DC 115	Engineering Change Process	14
EN-DC-112	Engineering Change Request and Project Initiation Process	5
EN-LI-009	10 CFR 50.59 Evaluations	9
EN-LI-100	Process Applicability Determination	13
OP-001-002	Reactor Coolant Pump Operation	21
OP-009-005	Shutdown Cooling System	31
OP-009-008	Safety Injection System	33
OP-010-003	Plant Startup	325
OP-010-005	Plant Shutdown	320
OP-500-013	Control Room Cabinet SA	17
OP-500-014	Control Room Cabinet SB	18
OP-901-202	Steam Generator Tube Leakage or High Activity	10
OP-901-202	Steam Generator Tube Leakage or High Activity	11
OP-901-502	Evacuation of Control Room and Subsequent Plant Shutdown	23
OP-901-502	Evacuation of Control Room and Subsequent Plant Shutdown	26
OP-901-511	Instrument Air Malfunction	11
OP-901-513	Spent Fuel Pool Cooling Malfunction	8
OP-902 008	Emergency Operating Procedure, Functional Recovery Procedure	20
OP-902-002	Loss of Coolant Accident Recovery	17
OP-903-011	High Pressure Safety Injection Pump Pressure Operability Check	12
OP-903-026	Emergency Core Cooling System Valve Lineup Verification	22
OP-903-033	Cold Shutdown IST Valve Tests	39
OP-903-035	Containment Spray Pump Operability Check	18
OP-903-115	Train “A” Integrated Emergency Diesel Generator/Engineering Safety features Test,	24
SEP-WF3-IST-1	WF3 In-service Testing Bases Document	01
STA-001-004	Local Leak Rate Test	309
UNT-007-060	Control of Loose Items	303

PROCEDURES

NUMBER	TITLE	REVISION
W3-DBD-001	Safety Injection System	302
W3-DBD-002	Emergency Diesel Generator & Automatic Load Sequence	13
W3-DBD-013	Containment Spray System	301

50.59 SCREENS ASSOCIATED WITH ENGINEERING CHANGES

PROCEDURES

NUMBER	TITLE	REVISION
PAD EC-02188	Perform The Chilled Water Design Change Described in CR-WF3-2006-00440	0
PAD EC-07436	CPC Constant Update	0
PAD EC-08428	Miscellaneous Structural Inside Containment for the SG/RVCH	0
PAD EC-10454	Modify Upper Guide Structure Lift Rig to Support ICI Thimble	0
PAD EC-12084	TS Bases 3/4.3.1-3/4.3.2 4KV Safety Bus UV Relay Clarification	0
PAD EC-12592	Installation of Conduit In Containment to Facilitate RCP Seal Inspection	0
PAD EC-17647	Agastat Relay Replacement	0
PAD EC-18232	Fuel Pool Pumps A&B Water Level Switch Bypass,	1
PAD EC-24060	Incorporate RG 1.121 Evaluation	0
PAD EC-26496	Technical Specification 3.4.1.5 Operability Evaluation	0
PAD EC-30731	Add Limit Trip Switch Plate to Cask Crane	0
PAD EC-30976	Backup Instrument Air Supply for SI-129A(B) to Throttle	0
PAD EC-31375	Replace existing 15 Inch Diameter Hand Wheel On CS-117A (B) with 60 Inch Diameter Hand Wheel With Torque Limiter Set 237 Ft-1b	0
PAD EC-31826	Replays Obsolete Emergency Diesel Generator (EDG) Sequencer Relays (Train A)	0
PAD EC-41604	Reactor Coolant P/T Curves	0
PAD EC-42373	Add Vibration Dampener To Pipe Between Pipe Cap And Valve FW-1651B and Valve FW-1681B	0
PAD EC-42374	Add Vibration Dampener To Pipe Below FW-189B	0

PAD EC-84913	RAB Room Area Temperature Monitoring Instrumentation Ranges	0

ENGINEER CHANGES

NUMBER	TITLE	REVISION
EC-02188	Perform the Chilled Water Design Change Described in Cr-Wf3-2006-00440	0
EC-07436	CPC Constraint Update	0
EC-08428	Miscellaneous Structural Inside Containment for the SG/RVCH	0
EC-10454	Modify Upper Guide Structure Lift Rig to Support ICI Thimble	0
EC-12084	TS Bases 3/4.3.1-3/4.3.2 4Kv Safety BUS UV Relay Clarification	0
EC-12592	Installation Of Conduit In Containment to Facilitate RCP Seal Inspection	0
EC-17647	Install Replacement Timer Relays in Aux Panels 1C and 2C	0
EC-18232	Fuel Pool Pumps A & B Water Level Switch Bypass and 5059 Evaluation	1
EC-24060	Incorporate RG 1.121 Evaluation	0
EC-25944	Changes to EC-14765 for CALC MPR-2390	0
EC-26496	Technical Specification 3.4.1.5 Operability Evaluation	
EC-30731	Add Limit Trip Switch Plate to Cask Crane	0
EC-30976	Backup Instrument Air Supply for SI-129A(B) to Throttle	0
EC-31375	Replace Existing 15 Inch Diameter Hand Wheel on CS-117A (B) with 60 inch Diameter Hand Wheel with Torque Limiter Set 237 Ft-lb.	0
EC-31826	Replays Obsolete Emergency Diesel Generator (EDG) Sequencer Relays (Train A)	0
EC-41604	Reactor Coolant P/T Curves	0
EC-42373	Add Vibration Dampener To Pipe Between Pipe Cap and Valve FW-1651B and Valve FW-1681B	0
EC-42373	Add Vibration Dampener to Pipe Below FW-189B	0
EC-5000084913	RAB Room Area Temperature Monitoring Instrumentation Ranges	0

MISCELLANEOUS

NUMBER	TITLE	REVISION
W3F1-2005-0005	90 day Response to Generic Letter 2004-02	March 3, 2005
W3F1-2005-0052	Response to Request For Additional Information for Generic Letter 2004-02	July 28, 2005
W3F1-2005-0063	Response to Generic Letter 2004-02	Sept. 16. 2005
W3F1-2006-0038	Generic Letter 2004-02 Commitment Revision	Nov. 29, 2006
W3F1-2010-0032	Response to Request for Additional Information Regarding Final Supplemental Response to Generic Letter 2004-02	November 23, 2010
W3F1-2005-0005	90 day Response to Generic Letter 2004-02	March 3, 2005
W3F1-2011-0012	Commitment Change for Resolution of Generic Letter 2004-02 Due to Deferral of Replacing of Waterford 3 Steam Generator	February 9, 2011

WORK ORDERS

WO 275977	WO 198950	WO 212017	WO 326332
WO 326333	WO 212017	WO 198950	

Section 1R18: Plant Modifications

PROCEDURES

NUMBER	TITLE	REVISION
EC-43821	Provided Regulated Nitrogen Supply to Safety Injection Tank 2B	0
EC-45104	The Check Valves Installed per Temporary Modification EC-43821 were Below the Design Pressure Rating of the SIT 2B Instrumentation	0

Section 1R19: Post-Maintenance Testing

PROCEDURES

NUMBER	TITLE	REVISION
OP-009-002	Emergency Diesel Generator	320
OP-009-002	Temporary Emergency Diesel Generator	1
OP-TEM-008	Emergency Diesel Generator A(B) Backup Temporary Diesel	5

Section 1R19: Post-Maintenance Testing

PROCEDURES

NUMBER	TITLE	REVISION
	Generator	
OP-903-068	Emergency Diesel Generator Operability & Subgroup Relay Operability Verification	307
OP-903-119	Secondary Auxiliaries Quarterly IST Valve Test	17
OP-002-010	Reactor Auxiliary Building HVAC & Containment Purge	306

CONDITION REPORTS

CR-WF3-2013-2035 CR-WF3-2013-2113 CR-WF3-2013-2946

WORK ORDERS

WO 00349445 WO 00194928 WO 00251396 WO 00348991

WO 00345809 WO 00353653

Section 1R20: Refueling and Other Outage Activities

PROCEDURES

NUMBER	TITLE	REVISION
OP-010-003	Plant Startup	327
OP-100-014	Technical Specifications and Technical Requirement Compliance	313
OP-903-001	Technical Specification Surveillance Logs	53
OP-903-090	Shutdown Margin Verification	303

CONDITION REPORTS

CR-WF3-2013-2166 CR-WF3-2013-3004 CR-WF3-2013-3182 CR-WF3-2013-2118

MISCELLANEOUS

NUMBER	TITLE	REVISION
W3F1-2002-0095	License Amendment Request for NPF-38-243, Reactivity / Boron Concentration Change	Oct. 24, 2002
	WSES, Unit 3 – Issuance of Amendment Re: Reactivity / Boron Concentration Changes (TAC NO. MB6616)	March 7, 2003

TSTF-286	Technical Specification Task Force – Define Operations Involving Positive Reactivity Additions	2

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION
OP-903-030	Safety Injection Pump Operability	19
OP-903-068	Emergency Diesel Generator Operability & Subgroup Relay Operability Verification	307
OP-903-024	Reactor Coolant System Water Inventory Balance	20
OP-903-128	Category A Leak Test	7
OP-903-123	Control Room Envelope Pressure Test	304
OP-002-003	Component Cooling Water	311
OP-903-050	Component Cooling Water and Auxiliary Component Cooling Water Pump and Valve Operability Test	28

CONDITION REPORTS

CR-WF3-2011-05159 CR-WF3-2011-06456 CR-WF3-2011-08024 CR-WF3-2012-00011

CR-WF3-2012-00022 CR-WF3-2012-00934 CR-WF3-201203054

WORK ORDERS

WO 52479331

Section 1EP6: Drill Evaluation

PROCEDURES

NUMBER	TITLE	REVISION
EP	Emergency Plan	43
EP-001-001	Recognition and Classification of Emergency Conditions	30
2013-02	Green Team Site Drill	June 12, 2013

CONDITION REPORTS

CR-WF3-2013-2875 CR-WF3-2013-2883 CR-WF3-2013-3202 CR-WF3-2013-3238

CR-WF3-2013-3275

Section 4OA1: Performance Indicator Verification

PROCEDURES

NUMBER	TITLE	REVISION
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	6
EN-LI-114	Performance Indicator Process	4

Section 4OA2: Identification and Resolution of Problems

PROCEDURES

NUMBER	TITLE	REVISION
EN-LI-102	Corrective Action Process	20
EN-LI-119	Apparent Cause Evaluation (ACE) Process	16
EN-LI-118	Root Cause Evaluation Process	18

CONDITION REPORTS

CR-WF3-2013-2564	CR-WF3-2012-6703	CR-WF3-2012-7566	CR-WF3-2012-7570
CR-WF3-2012-7568	CR-WF3-2012-6545		

MISCELLANEOUS

NUMBER	TITLE	REVISION
PSA-WF3-08-011	WF3 Air Operated Valve Handwheel SDP Report	0
W3F1-91-0434	GL 88-14 Response, Instrument Air Supply System Problems Affecting Safety-Related Equipment	0

Section 4OA3: Event Followup

PROCEDURES

NUMBER	TITLE	REVISION
LER 05000382/2013-001-00	Main Feedwater Regulating Valve Closure Results in Automatic Reactor Trip	0

Section 4OA5: Other Activities

PROCEDURES

NUMBER	TITLE	REVISION
ECT-8458-001	Attachment 11-8, RSG Vibration Monitoring Plan	1

Section 4OA5: Other Activities

PROCEDURES

NUMBER	TITLE	REVISION
EN-OP-111	Operational Decision Making Issue	9
EN-DC-343	Underground Piping and Tanks Inspection and Monitoring Program	8
CEP-UPT-0100	Underground Piping and Tanks Inspection and Monitoring	2
SEP-UIP-WF3	Underground Components Inspection Plan	1

CONDITION REPORTS

CR-WF3-2013-0445 CR-WF3-2013-0453 CR-WF3-2013-0565 CR-WF3-2013-0527

CALCULATIONS

NUMBER	TITLE	REVISION
EC-8460	Replacement Steam Generator (RSG) Design / Qualification Electrical / I&C Systems and Components Impact Review	0
EC-42257	FW173A Actuator Accessory and Tubing Reroute	1

WORK ORDERS

WO 293343 WO 182700 WO 336725 WO 234047

MISCELLANEOUS

NUMBER	TITLE	DATE/REVISION
Engineering Project No. 1346	Evaluation of Fractured Lower Supply Air Line of 173A Actuator	Jan 22, 2013
ECT 8458	Replacement Steam Generator Vibration Data Measurements	1
ANSI/ASME OM3	Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems	1982
	Feedwater Piping Vibration Monitoring Plan	Jan 28, 2013
NEI 09-14	Guideline for the Management of Underground Piping and Tank Integrity	1