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

July 30, 2015

Louis P. Cortopassi, Site Vice President
Omaha Public Power District
Fort Calhoun Station
P.O. Box 550
Fort Calhoun, NE 68023-0550

Subject: FORT CALHOUN - NRC INTEGRATED INSPECTION REPORT NUMBER 05000285/2015002

Dear Mr. Cortopassi:

On June 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station (FCS). On July 24, 2015, the NRC inspectors discussed the results of this inspection with you, and other members of your staff. The inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented eight findings of very low safety significance (Green) in this report. Seven of these findings involved violations of NRC requirements. Additionally, NRC inspectors documented one Severity Level IV violation with no associated finding. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the NRC Enforcement Policy.

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

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement 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 the Fort Calhoun Station.

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

Document Room or from the Publicly Available Records (PARS) component of the NRC's Agency wide Documents 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/

Geoffrey B. Miller
Chief, Project Branch D
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2015002
w/Attachment: Supplemental Information

cc w/ encl: Electronic Distribution

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Letter to L. Cortopassi from Geoffrey B. Miller, dated July 30, 2015

Subject: FORT CALHOUN - NRC INTEGRATED INSPECTION REPORT NUMBER 05000285/2015002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2015002
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	April 1 through June 30, 2015
Inspectors:	S. Schneider, Senior Resident Inspector B. Cummings, Resident Inspector L. Carson, Senior Health Physicist P. Elkmann, Senior Emergency Preparedness Inspector P. Hernandez, Health Physicist I. Anchondo, Reactor Inspector M. Baquera, Materials Inspector B. Hagar, Senior Project Engineer
Approved By:	Geoffrey B. Miller, Chief, Projects Branch D Division of Reactor Projects

SUMMARY

IR 05000285/2015002; 4/01/2015 – 6/30/2015; Fort Calhoun Station, Inservice Inspection Activities, EAL and Emergency Plan Changes, Follow-up of Events

The inspection activities described in this report were performed between April 1 and June 30, 2015, by the resident inspectors at FCS, inspectors from the NRC's Region IV office, and inspectors from the office of Nuclear Regulatory Research (RES). Eight findings of very low safety significance (Green) are documented in this report. Seven of these findings involved violations of NRC requirements. Additionally, NRC inspectors documented in this report one Severity Level IV violation with no associated finding. The significance of inspection findings is indicated by their color (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process." Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas." Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process."

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation of 10 CFR 50.55a(g)(4), involving the failure to adequately perform periodic reactor coolant system (RCS) integrity inspections as required by ASME Code Section XI. Specifically, Procedure OP-ST-RC-3007, "Periodic Reactor Coolant System Integrity Test," required testing of all ASME Class 1 pressure boundary components of the reactor vessel pressure boundary but failed to include reactor vessel head vent line RC-2501R. As a result, the requirements of ASME Code Section XI were not met. This issue was entered into the licensee's corrective action program as Condition Report 2015-05858.

  The inspectors concluded that the failure to include reactor vessel head vent line RC-2501R within the reactor vessel pressure boundary in the periodic RCS integrity inspection was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the procedure quality 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 shutdown as well as power operations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not result in exceeding the RCS leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined that the finding had a conservative bias cross-cutting aspect in the area of human performance because the licensee failed to use decision making-practices that emphasized prudent choices over those that are simply allowable. [H.14] (Section 1R08)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to establish adequate work instructions to clean and inspect the reactor vessel head. Specifically, the work instructions for the required visual examination of the reactor vessel head failed to specify what constituted a relevant condition as defined by ASME Code Case N-729-1, "Alternative

Examination Requirements for PWR Reactor Vessel Upper Head with Nozzles Having Pressure Retaining Partial Penetration Welds." As a result, the licensee failed to identify several relevant conditions that required additional inspections to adequately assure that the structural integrity of the reactor vessel head was not compromised. This issue was entered into the licensee's corrective action program as Condition Report 2015-05995.

The failure to establish adequate work instructions to clean and inspect the reactor vessel head was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the procedure quality 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 shutdown as well as power operations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not result in exceeding the RCS leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined that the finding had a teamwork cross-cutting aspect in the area of human performance because individuals and work groups failed to communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety is maintained. [H.4] (Section 1R08)

- Green. The inspectors identified a non-cited violation of Technical Specification 5.8.1.a associated with the failure to establish a preventative maintenance schedule for the reactor vessel head vent manual isolation valve, RC-100. Specifically, engineering personnel failed to consider vendor recommended maintenance activities/schedules, and determined that the valve could be run to failure. As a result, when the valve packing failed during operation, boric acid leaked onto the reactor vessel head. The licensee replaced the valve internals during refueling outage RFO 27 under Work Order 551054. This issue was entered into the licensee's corrective action program as Condition Report 2015-05432.

  The failure of engineering personnel to establish a preventative maintenance schedule for the reactor vessel head vent manual isolation valve was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the equipment performance 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 shutdown as well as power operations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not result in exceeding the RCS leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. No cross-cutting aspect was assigned because the inspectors determined that the finding was not indicative of current performance. (Section 1R08)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to identify and correct a condition adverse to quality. Specifically, maintenance personnel failed to document a loose connection on incore instrument port 44 in the corrective action program. As a result, the connection was not tightened and boric acid leaked onto the reactor vessel head during operation. This

issue was entered into the licensee's corrective action program as Condition Report 2015-05864.

The failure of maintenance personnel to document a loose connection on incore instrument port 44 in the corrective action program was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the equipment performance 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 shutdown as well as power operations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not result in exceeding the RCS leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined that the finding had a field presence cross-cutting aspect in the area of human performance because the licensee did not ensure supervisory and management oversight of work activities, including contractors and supplemental personnel. [H.2] (Section 1R08)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of very low safety significance of 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures, and Drawings," because activities affecting quality were not accomplished in accordance with instructions and procedures established by the licensee. Specifically, the licensee failed to document a degraded condition associated with a safety related seismic snubber affecting the auxiliary feedwater system, did not notify operations of the degraded condition, and did not assess the risk of the inoperable snubber in accordance with licensee instructions and procedures. The licensee entered this violation into their corrective action program. Immediate actions taken to address this violation included a review of all other snubber inspections that were rejected to ensure that other degraded conditions were reported to the control room, a review of all planned snubber maintenance with respect to online risk, and the issuance of interim guidance to all Shift Managers on the subject of snubber operability and risk.

  The inspectors determined that the licensee's failure to follow instructions and procedures associated with safety related snubbers was a performance deficiency. The finding is more than minor because if left uncorrected, the performance deficiency could have led to a more significant safety concern. Specifically, the failure to follow instructions and procedures associated with safety related snubbers could result in unacceptable risk configurations that are not analyzed under technical specifications and could challenge the reliability of safety related equipment during a seismic event. Using NRC Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power", Exhibit 2 "Mitigating System Screening Questions" Part B, dated July 1, 2012, the inspectors determined the finding to be of very low safety significance (Green) since the finding did not result in the loss of equipment specifically designed to mitigate a seismic initiating event. The finding has a cross-cutting aspect in the area of Human Performance, the Work Management aspect, since the licensee did not implement a work process that ensured the identification and management of risk commensurate to the work. [H.5] (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of 10 CFR 50.55a(g)(4) for the failure to perform a valid 40-month inservice test of the spent fuel pool cooling system. Specifically, the licensee failed to identify an existing through-wall leak on discharge header vent valve AC-898 that invalidated the test. The licensee replaced vent valve AC-898 and repaired the affected weld in April 2015. This issue was entered into the licensee's corrective action program as Condition Report 2015-05038.

  The failure to perform a valid 40-month inservice test of the spent fuel pool cooling system was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the design 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. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 3, "Barrier Integrity Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the SFP that caused mechanical damage to fuel clad and a detectible release of radionuclides, did not result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and did not affect the SFP neutron absorber, fuel bundle misplacement or soluble boron concentration. The inspectors determined that the finding had a conservative bias cross-cutting aspect in the area of human performance because individuals failed to use decision making-practices that emphasized prudent choices over those that are simply allowable. Although the licensee had previously identified the leak in valve AC-898 and determined that the leak had compromised the structural integrity of the system, the licensee failed to fix the leak. [H.14] (Section 1R08)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to promptly identify and correct a condition adverse to quality. Specifically, the licensee failed to take corrective action to replace spent fuel pool cooling system discharge header vent valve AC-898 after a leak was identified. A work order for the condition was opened in 2009 but was never implemented. Subsequently, a pressure boundary leak was identified in 2013 and misidentified in 2014 but was never addressed. The licensee replaced vent valve AC-898 and repaired the affected weld in April 2015. This issue was entered into the licensee's corrective action program as Condition Report 2015-05038.

  The failure to promptly identify and correct a condition adverse to quality was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the design 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. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 3, "Barrier Integrity Screening Questions," the issue screened as having very low safety significance (Green) because the

finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the SFP that caused mechanical damage to fuel clad and a detectible release of radionuclides, did not result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and did not affect the SFP neutron absorber, fuel bundle misplacement or soluble boron concentration. The inspectors determined that the finding had a basis for decision cross-cutting aspect in the area of human performance because leaders failed to ensure that the bases for operational and organizational decisions were communicated during multiple instances where the leak in valve AC-898 could have been repaired. [H.10] (Section 1R08)

- Green. The inspectors identified a finding associated with the failure of operations personnel to follow procedures used to perform functionality assessments. Specifically, operations personnel failed to provide sufficient technical justification for the reasonable assurance of functionality of the spent fuel pool cooling system when boric acid leaks were identified on discharge header vent valve AC-898. Vent valve AC-898 was replaced and the issue was entered into the licensee's corrective action program as Condition Report 2015-05856.

  The failure of operations personnel to follow station procedures to perform functionality assessments was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the design 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. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 3, "Barrier Integrity Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the SFP that caused mechanical damage to fuel clad and a detectible release of radionuclides, did not result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and did not affect the SFP neutron absorber, fuel bundle misplacement or soluble boron concentration. The inspectors determined that the finding had a training cross-cutting aspect in the area of human performance because the licensee did not provide training and ensure knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values. [H.9] (Section 1R08)

Cornerstone: Emergency Preparedness

- SL-IV. The inspector identified a non-cited violation of 10 CFR 50.54(q)(5) for the licensee's failure to submit reports of its analysis of the impact of changes to the emergency plan and implementing procedures on the emergency plan. Specifically, the inspector identified three examples between February 21 and June 18, 2015, of the licensee submitting changes to the emergency plan and implementing procedures without the required

summaries. The issue was entered into the licensee's corrective action program as Condition Report CR 2015-04934.

The failure to submit a summary of the analysis of the effect of changes to emergency plan implementing procedures on the site emergency plan is a performance deficiency within the licensee's ability to foresee and correct. The issue is more than minor because the licensee's failure to submit the required summary affects the NRC's ability to perform its regulatory function, and the licensee has not incorporated this requirement into its program. The inspectors evaluated the issue using Section 6.6.d of the NRC Enforcement Policy, dated July 12, 2011, and determined it to be a Severity Level IV violation because the issue involved the licensee's ability to implement a regulatory requirement not related to assessment or notification. Traditional enforcement violations are not assigned a cross-cutting aspect. (Section 1EP4)

PLANT STATUS

The unit began the inspection period at approximately 100 percent power. On April 11, 2015, the unit was taken offline for a scheduled refueling outage. On June 10, 2015, the unit returned online and reached 100 percent power on June 14, 2015. On June 22, 2015, the unit was downpowered to 60 percent power to make repairs to electrical bus cooling systems that support the main transformer and returned to 100 percent power on June 23, 2015. On June 27, 2015, the unit was downpowered to approximately 98 percent power for moderator temperature coefficient testing. The unit achieved approximately 100 percent power on June 28, 2015, and operated at that power level 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)

Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

On April 1, 2015, the inspectors completed an inspection of the station’s readiness for impending adverse weather conditions. The inspectors reviewed plant design features, the licensee’s procedures to respond to forecasted high winds and severe weather, and the licensee’s implementation of these procedures. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant.

This activity constituted one sample of readiness for impending adverse weather conditions, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown

a. Inspection Scope

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

- April 15, 2015, engineered safeguards controls during cold shutdown operations
- April 21, 2015, spent fuel pool cooling system prior to core offload
- April 28, 2015, engineered safeguards controls while defueled

- May 7, 2015, raw water/component cooling water interface valves following inservice testing

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

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

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

On May 25, 2015, the inspectors performed a complete system walk-down inspection of the shutdown cooling system. The inspectors reviewed the licensee's procedures and system design information to determine the correct shutdown cooling system lineup for the existing plant configuration. The inspectors also reviewed outstanding work orders, open condition reports, and other open items tracked by the licensee's operations and engineering departments. The inspectors then visually verified that the system was correctly aligned for the existing plant configuration.

These activities constituted one complete system walk-down sample, as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

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

- April 3, 2015, intake structure, fire area 31 and east and west switchgear rooms, fire areas 36A and 36B for fire risk management actions during maintenance on raw water pump AC-10D

- April 21, 2015, containment, reactor coolant pump "B" bay, fire area 30

- April 23, 2015, containment, reactor coolant pump "A" bay, fire area 30

- May 4, 2015, room 5, spent fuel pool cooling pump and heat exchanger area, fire area 68

- June 16, 2015, room 59, pipe penetration, fire area 23

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

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

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On April 17, 2015, the inspectors completed an inspection of the station's ability to mitigate flooding due to internal causes. After reviewing the licensee's flooding analysis, and in consideration of raw water system maintenance, the inspectors chose one plant area containing risk-significant structures, systems, and components that were susceptible to flooding:

- Room 19, compressor area and auxiliary feedwater pump room, during maintenance on the east raw water header

The inspectors reviewed plant design features, the raw water maintenance isolation boundaries, and licensee procedures for coping with internal flooding. The inspectors walked down the selected areas to inspect the design features, including the material condition of seals, drains, and flood barriers.

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

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

On April 24, 2015, the inspectors completed an inspection of the readiness and availability of risk-significant heat exchangers. The inspectors reviewed the data from a performance test for the Shutdown Cooling Heat Exchanger A and verified the licensee used the industry standard periodic maintenance method outlined in EPRI NP-7552.

Additionally, the inspectors walked down Shutdown Cooling Heat Exchanger A to observe its performance and material condition and to verify that the heat exchanger was correctly categorized under the Maintenance Rule and was receiving the required maintenance.

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

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

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

.1 Non-destructive Examination (NDE) Activities and Welding Activities

a. Inspection Scope

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Low Pressure Safety Injection	Tee to Pipe (Weld No. 10)	Ultrasonic
Low Pressure Safety Injection	Pipe to Tee (Weld No. 11)	Ultrasonic
Shutdown Cooling	Valve Body (HCV-347)	Liquid Penetrant

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Safety Injection	6-SI-14 (Weld No. 12)	Ultrasonic
Safety Injection	6-SI-14 (Weld No. 13)	Ultrasonic
High Pressure Injection	3-HPH-14 (Weld Nos. 6,8,9)	Liquid Penetrant
Reactor Coolant System	Loop 1 Hotleg Nozzle (MRC-1/01)	Ultrasonic
Reactor Coolant System	Loop 2 Hotleg Nozzle (MRC-2/01)	Ultrasonic
Safety Injection	C.S. Pump SI-3B/Pump B (Weld Nos. F-14A, F-14D, F-16R1)	Radiograph
Safety Injection	C.S. Pump SI-3C/Pump C	Radiograph

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
	(Weld Nos. F-5A, F-5D)	
Safety Injection	LPSI Pump 3-B (Weld Nos. 16, 16B)	Radiograph

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

b. Findings

(1) Failure to Include a Class 1 Component in the Reactor Vessel Pressure Boundary Integrity Test

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR 50.55a(g)(4) involving the failure to adequately perform periodic reactor coolant system (RCS) integrity inspections as required by ASME Code Section XI. Specifically, procedure OP-ST-RC-3007, "Periodic Reactor Coolant System Integrity Test," required testing of all ASME Class 1 pressure boundary components of the reactor vessel pressure boundary but failed to include reactor vessel head vent line RC-2501R. As a result, the requirements of ASME Code Section XI were not met.

Description. During refueling outage RFO 27, the licensee discovered a packing leak on reactor vessel vent isolation valve, RC-100. The valve is located near the reactor vessel head insulation package and resulted in a leak onto the vessel head. Reactor vessel vent isolation valve RC-100 is part of vent line RC-2501R, and is classified as an ASME Class 1 system. ASME Code Section XI, Subarticle IWB-2500, "Examination and Pressure Test Requirements," for Class 1 systems requires that components be examined and pressure tested as specified in Table IWB-2500-1. Vent line RC-2501R is classified as examination category "B-P, All Pressure Retaining Components," item number B15.10; according to Table IWB-2500-1, it requires a system leakage test (VT-2) during each refueling outage.

The inspectors reviewed past system leakage test records of the reactor vessel head pressure retaining components as performed under licensee procedure OP-ST-RC-3007, "Periodic Reactor Coolant System Integrity Test." The inspectors determined that this procedure was included in the licensee's inservice inspection program and was implemented to satisfy the ASME Code Section XI requirements for Class 1 components, i.e., examination category "B-P, All Pressure Retaining Components." As such, the licensee should have included vent line RC-2501R (including valve RC-100) in procedure OP-ST-RC-3007. However, the inspectors identified that vent line RC-2501R was not included in that procedure.

The valve was replaced in 2006. Although no other leaks had been identified prior to RFO 27, the inspectors concluded that the licensee had not been inspecting the vent line and valve since at least the date of the valve's replacement in 2006. Had the licensee inspected vent line RC-2501R during the system leakage test of the reactor vessel

pressure boundary, they likely would have identified that the packing on vent valve RC-100 had degraded.

The licensee replaced the yoke assembly of vent valve RC-100 during RFO 27 under Work Order 551054. The licensee documented this issue in the corrective action program as Condition Report 2015-05858.

Analysis. The inspectors concluded that the failure to include vent line RC-2501R within the reactor vessel pressure boundary in the periodic RCS integrity inspection was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the procedure quality 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 shutdown as well as power operations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not result in exceeding the RCS leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined that the finding had a conservative bias cross-cutting aspect in the area of human performance because the licensee failed to use decision making-practices that emphasized prudent choices over those that are simply allowable. Specifically, licensee personnel performed this inspection following every outage but failed to use their knowledge of the reactor coolant system to question why the reactor vessel head vent line RC-2501R was not part of the pressure test. [H.14]

Enforcement. Title 10 CFR Part 50.55a(g)(4) states, in part, that throughout the service life of a pressurized water-cooled nuclear power facility, components which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements set forth in Section XI of editions and addenda of the ASME B&PV Code. ASME Code Section XI, Subarticle IWB-2500, "Examination and Pressure Test Requirements," states, in part, that components shall be examined and pressure tested as specified in Table IWB 2500-1. ASME Code Section XI, Table IWB-2500-1, "(B-P) Examination Category B-P, All Pressure Retaining Components," Item B15.10, specifies, in part, that a system leakage test on pressure retaining components must be performed during each refueling outage. Contrary to the above, since 2006, the licensee failed to perform an adequate system leakage test on pressure retaining components as required by ASME Code Section XI. Specifically, the licensee failed to inspect RVH vent line RC-2501R, including vent valve RC-100, during the system leakage test of the reactor vessel pressure boundary. Consequently, the licensee failed to identify that vent valve RC-100 had degraded valve packing.

In response to the packing leak, the licensee replaced the yoke assembly of vent valve RC-100. This finding was entered into the licensee's corrective action program as Condition Report 2015-05858. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015002-001 "Failure to Include a Class 1 Component in the Reactor Vessel Pressure Boundary Integrity Test."

(2) Failure to Perform a Valid 40-Month Inservice Test

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR 50.55a(g)(4) for the failure to perform a valid 40-month inservice test of the spent fuel pool cooling system. Specifically, the licensee failed to identify an existing through-wall leak on discharge header vent valve AC-898 that invalidated the test. The licensee replaced vent valve AC-898 and repaired the affected weld in April 2015.

Description. The spent fuel pool cooling system is classified as an ASME Class 3 system that requires a system leakage test every inspection period per the requirements of Section XI of the ASME Code. For Class 3 systems, Subarticle IWD-2500, "Examination and Pressure Test Requirements," states, in part, components shall be examined and pressure tested as specified in Table IWD-2500-1. Table IWD-2500-1 specifies, in part, that a system leakage test on pressure retaining components must be performed during each inspection period. The inspectors determined that the licensee implemented this Code requirement using Procedure QC-ST-SFP-3001, "Forty-Month Inservice Test of the Spent Fuel Pool Cooling System in Room No. 5."

Valve AC-898 is the discharge header vent valve for spent fuel pool circulation pumps 5A and 5B. On July 30, 2013, the licensee identified that vent valve AC-898 had a pinhole leak at the valve to header pipe weld. This issue was documented in Condition Report 2013-15233 and subsequently closed to Work Request 199765. The inspectors determined that Work Request 199765 was never implemented.

On December 26, 2014, the licensee performed a 40-month inservice test of the spent fuel pool cooling system in accordance with Procedure QC-ST-SFP-3001. During the test, the licensee documented that valve AC-898 had a packing leak. Because the packing leak was not considered a pressure boundary leak, the licensee recorded this test as satisfactory. However, the inspectors found that Procedure QC-ST-SFP-3001, Step 10.1, stated that "Any through wall leakage of a pressure retaining weld, fitting, piping, pump casing or valve body/bonnet constitutes a failure of the test." The inspectors concluded that the failure to address the pinhole leak in 2013 and subsequently misidentifying this pressure boundary leakage as a packing leakresulted in an invalid 40-month inservice test. As corrective actions, the licensee replaced vent valve AC-898 and repaired the affected weld in April 2015 under Work Order 550470. This issue was entered into the corrective action program as Condition Report 2015-05862.

Analysis. The inspectors concluded that the failure to perform a valid 40-month inservice test of the spent fuel pool cooling system was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the design 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. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 3, "Barrier Integrity Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over

the spent fuel pool that caused mechanical damage to fuel clad and a detectible release of radionuclides, did not result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and did not affect the spent fuel pool neutron absorber, fuel bundle misplacement or soluble boron concentration. The inspectors determined that the finding had a conservative bias cross-cutting aspect in the area of human performance because individuals failed to use decision making-practices that emphasized prudent choices over those that were simply allowable. Although the licensee had previously identified the leak in valve RC-898 and determined that the leak had compromised the structural integrity of the system, the staff failed to fix the leak. [H.14]

Enforcement. Title 10 CFR 50.55a(g)(4) states, in part, that throughout the service life of a pressurized water-cooled nuclear power facility, components which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements set forth in Section XI of editions and addenda of the ASME B&PV Code. ASME Code Section XI, Subarticle IWD-2500, "Examination and Pressure Test Requirements," states, in part, that components shall be examined and pressure tested as specified in Table IWD-2500-1. ASME Code Section XI, Table IWD-2500-1, "(D-B) Examination Category D-B, All Pressure Retaining Components," specifies, in part, that a system leakage test on pressure retaining components must be performed during each inspection period. Contrary to the above, on December 26, 2014, the licensee failed to perform a valid system leakage test on pressure retaining components. Specifically, the licensee failed to repair vent valve AC-898 in 2013 and subsequently failed to appropriately identify this leak during the performance of the 40-month inservice test of the spent fuel pool system as required by Section XI of the ASME Code.

In response to the leak, the licensee replaced vent valve AC-898 and repaired the affected weld. This finding was entered into the licensee's corrective action program as Condition Report CR 2015-05862. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015002-002 "Failure to Perform a Valid 40-Month Inservice Test."

.2 Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

During the bare metal visual inspection, the licensee identified what appeared to be boric acid on top of the reactor vessel upper head. As a result, the licensee was required to take additional actions to address this condition as required by ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Head with Nozzles Having Pressure Retaining Partial Penetration Welds." However, the licensee failed to inspect a number of nozzles in accordance with Code Case N-729-1 prior to cleaning the reactor vessel head. These actions place the licensee in noncompliance with Code Case N-729-1 and thereby 10 CFR 50.55a(g)(6)(ii)(D). The licensee submitted Relief Request RR-14, "Alternate Inspection of Reactor Vessel Closure Head Penetration Nozzles" by letter dated May 9, 2015, supplemented by letters dated May 13, 16, and 17, 2015. The NRC granted verbal approval on May 17, 2015. At the time of the inspection, the NRC had not yet written a safety evaluation. The

licensee documented the condition on the reactor vessel head in Condition Report 2015-05255.

b. Findings

Failure to Establish Adequate Work Instructions to Clean and Inspect the Reactor Vessel Head

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to establish adequate work instructions to clean and inspect the reactor vessel head. Specifically, the work instructions for the required visual examination of the reactor vessel head failed to specify what constituted a relevant condition as defined by ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Head with Nozzles Having Pressure Retaining Partial Penetration Welds." As a result, the licensee failed to identify several relevant conditions that required additional inspections to adequately assure that the structural integrity of the reactor vessel head was not compromised.

Description. During refueling outage RFO 27, the licensee performed the required visual examination (VE) of the reactor vessel head as specified by ASME Code Case N-729-1. The requirements of Code Case N-729-1 have been incorporated into 10 CFR 50.55a(g)(6)(ii)(D), "Reactor Vessel Head Inspections."

During the initial VE inspection, the licensee discovered a significant amount of material, including boric acid deposits, on top of the reactor vessel head. This condition prompted the licensee to take additional actions beyond the VE inspection, as required by Code Case N-729-1. Specifically, the licensee was required to identify and select the nozzles that exhibited relevant conditions for further examinations. Code Case N-729-1 defines a relevant condition as:

> Relevant conditions for the purposes of the VE shall include areas of corrosion, boric acid deposits, discoloration, and other evidence of nozzle leakage.

Acceptance of components for continued service that exhibit relevant conditions is specified in subparagraph -3142.1(b), "Acceptance by VE," which states:

> A component whose VE detects a relevant condition shall be unacceptable for continued service until the requirements of -3142.1(b)(1), (b)(2), and (c) are met.

The licensee used the initial reactor vessel head inspection records to select the nozzles that exhibited relevant conditions. The licensee identified 24 of the 49 nozzles as exhibiting relevant conditions and thus warranting further examination. However, the inspectors reviewed the licensee data and identified additional nozzles with relevant conditions.

The licensee developed work instructions, "RC-6: Clean Boric Acid and Inspect Reactor Vessel Closure Head," under Work Order 551120-01 to perform the required additional inspections. Step 4.8 of RC-9 was written to describe the inspections required by Code Case N-729-1. Step 4.8 states,

Perform a VE inspection of the reactor vessel head for degradation. Record inspection data in accordance with all applicable QC inspection procedures. Use Attachment 2 to document your findings during the inspection.

The inspectors concluded that the work instructions did not specify what constituted a relevant condition and licensee staff did not refer to Code Case N-729-1 when selecting the nozzles of concern. As a result, the licensee failed to identify a number of nozzles that exhibited relevant conditions. Although the inspectors and NRR staff raised concerns that the proposed scope (i.e. nozzle selections) for the supplemental inspections did not meet the requirements of the Code Case, the licensee proceeded with performance of RC-6, including cleaning the reactor vessel head. As a result, the licensee could no longer perform the inspections that were missed. Subsequently, the licensee submitted Relief Request 14, "Alternate inspection of the Reactor Vessel Closure Head Penetration Nozzles"; the NRC granted verbal approval on May 17, 2015.

<u>Analysis</u>. The inspectors concluded that the failure to establish adequate work instructions to clean and inspect the reactor vessel head was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the procedure quality 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 shutdown as well as power operations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not result in exceeding the RCS leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined that the finding had a teamwork cross-cutting aspect in the area of human performance because individuals and work groups failed to communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety is maintained. Specifically, the individuals familiar with the requirements of the ASME Code and those developing and implementing the work instructions did not coordinate their activities and ensure the requirements of ASME Code Case N-729-1 were included in the work instructions. [H.4]

<u>Enforcement</u>. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished." Contrary to the above, on May 4, 2015, the licensee failed to establish appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to specify what constituted a relevant condition as defined by ASME Code Case N-729-1, and as a result, the licensee failed to inspect several nozzles that exhibited relevant conditions. In response to missing the required inspections, the licensee submitted Relief Request RR-14 and the NRC granted verbal approval of the licensee's submittal on May 17, 2015. Because this finding was of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2015-05995, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015002-003 "Failure to Establish Adequate Work Instructions to Clean and Inspect the Reactor Vessel Head."

.3 Boric Acid Corrosion Control (BACC) Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's implementation of its boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walk-down as specified in procedure ER-AP-331-1002, "Boric Acid Corrosion Control Program Identification, Screening, and Evaluation," Revision 8, and Procedure ER-AP-331-1001, "Boric Acid Corrosion Control (BACC) Inspection Locations, Implementation, and Inspection Guidelines," Revision 7. The inspectors reviewed whether the visual inspections emphasized locations where boric acid leaks could cause degradation of safety significant components, and whether engineering evaluation used corrosion rates applicable to the affected components and properly assessed the effects of corrosion induced wastage on structural or pressure boundary integrity. The inspectors observed whether corrective actions taken were consistent with the ASME Code and 10 CFR Part 50, Appendix B, requirements.

b. Findings

(1) Failure to Incorporate Vendor Manual Recommendations for Conducting Preventative Maintenance on the Reactor Vessel Head Vent Valve

Introduction. The inspectors identified a Green, non-cited violation of Technical Specification 5.8.1.a associated with the failure to establish a preventative maintenance schedule for the reactor vessel head vent manual isolation valve, RC-100. Specifically, engineering personnel failed to consider vendor recommended maintenance activities and schedules, determining that the valve could be run to failure. As a result, when the valve packing failed during operation, boric acid leaked onto the reactor vessel head.

Description. In July 2008, the licensee developed the preventative maintenance schedule for the reactor vessel head vent manual isolation valve RC-100. In this assessment, they categorized RC-100 as "run to failure". In contrast, the vendor manual for RC-100 established a periodicity of every 3 years for replacing the consumable valve packing. The licensee apparently did not consider this recommendation when determining the preventative maintenance schedule, and contrary to Procedure PED-SEI-13 "Preventative Maintenance Program – Technical Basis," Revision 12, the licensee did not provide a justification for deviating from the vendor recommendation. The failure of this valve would result in an unisolable leak of reactor coolant onto the reactor vessel head; consequently, the licensee should not have categorized it as "run to failure". The packing failed during operation and, as a result, the licensee found boric acid on the reactor vessel head during refueling outage RFO 27. The licensee replaced the valve internals during RFO 27 under Work Order 551054. This issue has been entered into the licensee's corrective action program as Condition Report 2015-05432.

Analysis. The failure of engineering personnel to establish a preventative maintenance schedule for reactor vessel head vent manual isolation valve RC-100 was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the equipment performance 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 shutdown as well as power operations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not result in exceeding the RCS leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. No cross-cutting aspect was assigned because the inspectors determined that the finding was not indicative of current performance.

Enforcement. Technical Specification 5.8.1.a requires, in part, that written procedures and administrative policies shall be established, implemented and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Revision 2, Appendix A, 1978, Section 9.b requires, in part that, preventive maintenance schedules should be developed to specify replacement of parts that have a specific lifetime. Contrary to the above, since July 2008, the licensee had not developed preventative maintenance schedules to specify replacement of parts that have a specific lifetime. Specifically, the licensee did not develop a preventative maintenance schedule for reactor vessel head vent manual isolation valve RC-100 to replace packing in the valve as specified by the vendor manual. As a result, the valve packing for RC-100 failed during the operation cycle, and boric acid leaked onto the reactor vessel head. The licensee replaced the valve internals during the refueling outage. Because this finding is of very low safety significance and the condition was entered into the licensee's corrective action program as Condition Report 2015-05342, this violation is being treated as a non-cited violation in accordance with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2015002-004 "Failure to Incorporate Vendor Manual Recommendations for Conducting Preventative Maintenance on the Reactor Vessel Head Vent Valve."

(2) Failure to Promptly Identify and Correct a Condition Adverse to Quality Involving a Spent Fuel Pool Cooling Vent Valve Leak

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to promptly identify and correct a condition adverse to quality. Specifically, the licensee failed to take corrective action to replace spent fuel pool cooling system discharge header vent valve AC-898 after a leak was identified. A work order for the condition was opened in 2009 but was never implemented. Subsequently, a pressure boundary leak was identified in 2013 and misidentified in 2014 but was never addressed.

Description. On January 28, 2009, the licensee identified that spent fuel pool discharge header vent valve AC-898 was leaking by the valve seat. Condition Report 2009-0408 was written to document the valve's condition. The condition report was subsequently closed to Work Order 331407, written to replace the valve. The inspectors determined that this work order was never implemented. The inspectors also identified that the licensee had failed to perform a boric acid evaluation documenting the condition of the valve at the time of discovery, as required by plant procedures.

On July 30, 2013, the licensee identified that vent valve AC-898 had a pinhole leak at the valve to header pipe weld. This issue was documented in Condition

Report 2013-15233 and subsequently closed to Work Request 199765. The inspectors identified that Work Request 199765 was never implemented. Vent valve AC-898 is classified as an ASME Class 3 component. However, the licensee failed to identify that this condition represented a challenge to the structural integrity of the pressure boundary of the SFP system and constituted a noncompliance with the requirements of ASME Code Section XI. Additionally, the licensee failed, for a second time, to perform a boric acid evaluation of the identified condition as required by plant procedures.

On December 26, 2014, the licensee documented in Condition Report 2014-15622 that vent valve AC-898 had a packing leak. This was identified during a 40-month inservice test of the SFP system performed to meet the requirements ASME Code Section XI. The inspectors concluded that the licensee improperly characterized the leak as a packing leak during the 40-month inservice test, in spite of having previously identified a pinhole leak in 2013 that had not been repaired.

The licensee replaced vent valve AC-898 and repaired the affected weld in April 2015 under Work Order 550470. This issue was entered into the corrective action program as Condition Report 2015-05038.

Analysis. The inspectors concluded that the failure to promptly identify and correct a condition adverse to quality was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the design 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. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 3, "Barrier Integrity Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the SFP that caused mechanical damage to fuel clad and a detectible release of radionuclides, did not result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and did not affect the SFP neutron absorber, fuel bundle misplacement or soluble boron concentration. The inspectors determined that the finding had a basis for decision cross-cutting aspect in the area of human performance because leaders failed to ensure that the bases for operational and organizational decisions were communicated during the multiple instances when the leak in valve RC-898 could have been repaired. [H.10]

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," states, in part, measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance are promptly identified and corrected. Contrary to the above, between January 28, 2009, and April 19, 2015, the licensee did not promptly identify and correct a condition adverse to quality. Specifically, the licensee failed to repair vent valve AC-898 in 2009, and subsequently failed to identify and repair a pinhole leak in the valve to header pipe weld in 2013 and 2014. In response to the leak, the licensee replaced vent valve AC-898 and repaired the affected weld. This finding was entered into the licensee's corrective action program as Condition

Report 2015-05038. Because this finding was of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2015002-005 "Failure to Promptly Identify and Correct a Condition Adverse to Quality Involving a Spent Fuel Pool Cooling Vent Valve Leak."

(3) Failure to Identify and Correct Loose Incore Instrument Nozzle Connection

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to identify and correct a condition adverse to quality. Specifically, maintenance personnel failed to document a loose connection on incore instrument port 44 in the corrective action program. As a result, the connection was not tightened and boric acid leaked onto the reactor vessel head during operation. This issue was entered into the licensee's corrective action program as Condition Report 2015-05864.

Description. In October 2013, while the licensee was restarting the plant from an extended outage, maintenance personnel performed Work Order 00390145 to reassemble Grayloc hubs on the incore instrument flanges. This work order required the torqueing of mechanical fasteners in accordance with Procedure PR-RR-ICI-0302, "Incore Instrument Bullet Nose Removal and Adapter Hub Installation," Revision 22. The inspectors noted that the procedure did not require documentation of what materials and testing equipment was used during the performance of this procedure, nor did the procedure require documentation of the as-left torque values. Through interviews with licensee personnel, the inspectors determined that the torque wrench used was in calibration; the inspectors also determined that the licensee used a separate database to cross reference which work order used what calibrated equipment. However, the inspectors determined that the licensee did not record as-left torque values. In the comments section of Work Order 00390145, maintenance personnel documented only that incore instrument port 44 had a loose connection and that it needed to be checked.

A review of the corrective action program identified that this condition had not been entered in the corrective action program or addressed prior to restart. Interviews with licensee staff also revealed that contractor personnel performed this work, under the supervision of the licensee. The licensee's maintenance personnel signed the work order as satisfactorily completed.

During refueling outage RFO 27 in 2015, the licensee found incore instrument port 44 to be leaking. At that time, the licensee obtained as-left torque values, determined incore instrument port 44 was loose, and found that two of the associated mechanical fasteners were damaged. They reworked the connection on incore instrument port 44 during RFO 27.

Analysis. The inspectors concluded the failure of maintenance personnel to document a loose connection on incore instrument port 44 in the corrective action program was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the equipment performance 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 shutdown as well as power operations. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings

At-Power," Exhibit 1, "Initiating Events Screening Questions," the issue screened as having very low safety significance (Green) because the finding did not result in exceeding the RCS leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined that the finding had a field presence cross-cutting aspect in the area of human performance because the licensee did not ensure supervisory and management oversight of work activities, including contractors and supplemental personnel. Specifically, although the loose connection had been documented by contractor staff in the comments section of the work order, licensee personnel signed- the work order as completed satisfactorily. [H.2].

Enforcement. Title 10 CFR Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, between October 2013 and May 5, 2015, the licensee did not promptly identify and correct a condition adverse to quality. Specifically, the licensee failed to document a loose connection on incore instrument port 44 in the corrective action program. As a result, the licensee did not tighten the connection and the connection subsequently leaked boric acid onto the reactor vessel head during the operating cycle. The connection on incore instrument port 44 was reworked during refueling outage RFO 27. Because this finding is of very low safety significance and the condition was entered into the licensee's corrective action program as Condition Report 2015-05864, this violation is being treated as a non-cited violation in accordance with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2015002-006 "Failure to Identify and Correct Loose Incore Instrument Nozzle Connection."

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed the steam generator tube eddy current (ECT) examination scope and expansion criteria to determine whether these criteria met technical specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors also reviewed whether the ECT inspection scope included areas of degradations that were known to represent potential eddy current test challenges such as the top of tube sheet, tube support plates, and U-bends. The inspectors confirmed that no repairs were required at the time of the inspection.

The licensee inspected 50 percent of the tubes in each steam generator and identified that tube support plate wear was the only degradation mechanism present. The largest indication constituted a 16 percent through-wall, well below the plugging limit. The inspectors observed portions of the eddy current testing being performed do determine whether: (1) the appropriate probes were used for identifying the expected types of degradation, (2) calibration requirements were followed, and (3) probe travel speed was in accordance with procedural requirements. The inspectors performed a review of the site-specific qualifications for the techniques being used and reviewed whether eddy current test data analyses were adequately performed per EPRI and site specific guidelines.

Finally, the inspectors reviewed selected eddy current test data to verify that the analytical techniques used were adequate.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of condition reports that dealt with inservice inspection activities and found the corrective actions for inservice inspection issues were appropriate. The specific condition reports reviewed are listed in the Attachment.

b. Findings

Failure to Perform Functionality Assessments for the Spent Fuel Pool Cooling System.

Introduction. The inspectors identified a Green finding for the failure of operations personnel to follow procedures used to perform functionality assessments. Specifically, operations personnel failed to provide sufficient technical justification for the reasonable assurance of functionality of the spent fuel pool cooling system when boric acid leaks were identified on discharge header vent valve AC-898.

Description. In January 2009, operations personnel identified a boric acid leak on the spent fuel pool cooling header vent valve AC-898, and documented this issue in Condition Report 2009-0408. Operations personnel performed a functionality assessment of the valve using Procedure OP-FC-108-115, "Operability Determinations." The inspectors identified that the assessment lacked sufficient technical justification to address the functionality of the valve and how the system would maintain functionality until repair of valve AC-898. The licensee had scheduled Work Order 331407 to replace the valve, but never implemented that work order.

In July 2013, the licensee documented valve AC-898 as having a pressure boundary leak in Condition Report 2013-15223. The operations personnel performed a functional assessment and determined that the spent fuel pool cooling system was functional but degraded, however they did not assess the continued degradation of the valve. The condition report was closed to Work Order 199765, which they had initiated to replace the valve.

In December 2014, the licensee determined valve AC-898 to be leaking during a system pressure test required by Procedure QC-ST-SFP-3001, "Forty-Month Inservice Test of the Spent Fuel Pool Cooling System in Room No. 5," and initiated a condition report. This condition report was also closed to Work Order 199765. The inspectors identified that operations personnel had failed to perform a functionality assessment of a degraded condition of valve AC-898.

In April 2015, maintenance personnel again identified valve AC-898 to be leaking. Operations personnel failed to perform a functionality assessment of a degraded condition as required by Procedure OP-FC-108-115.

Through discussions with operations personnel, inspectors determined that there was a lack of understanding on how to assess structural integrity issues within the functionality process. In addition, the inspectors found that the scope of examples included in the functionality template was so narrow that it could contribute to the failure to perform functionality assessments.

The licensee replaced valve AC-898 during refueling outage RFO 27, under Work Order 550470. The licensee entered this issue into their corrective action program as Condition Report 2015-05856.

Analysis. The inspectors concluded the failure of operations personnel to follow station procedures to perform functionality assessments was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the design 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. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” Exhibit 3, “Barrier Integrity Screening Questions,” the issue screened as having very low safety significance (Green) because the finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the spent fuel pool that caused mechanical damage to fuel clad and a detectible release of radionuclides, did not result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and did not affect the spent fuel pool neutron absorber, fuel bundle misplacement or soluble boron concentration. The inspectors determined that the finding had a training cross-cutting aspect in the area of human performance because the licensee did not provide training and ensure knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values. Specifically, training provided on performing operability determinations failed to adequately address the performance of functionality assessments. [H.9]

Enforcement. This finding does not involve enforcement action because no violation of a regulatory requirement was identified. Because this finding does not involve a violation and is of very low safety significance (Green), it is identified as FIN 05000285-21015002-007, “Failure to Perform Functionality Assessments for the Spent Fuel Pool Cooling System.”

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

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On May 18, 2015, the inspectors observed simulator training for an operating crew in preparation for a reactor start-up. The inspectors assessed the performance of the operators and the evaluators' critique of their performance. The inspectors also assessed the modeling and performance of the simulator during the training.

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 Review of Licensed Operator Performance

a. Inspection Scope

The inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity or risk. The inspectors observed the operators' performance of the following activities:

- May 23, 2015, operations at reduced reactor coolant system inventory while establishing conditions for vacuum fill. The inspectors observed the operators' performance related to the assessment of plant risk and the establishment of plant conditions required for the activity.

- On June 1, 2015, operations following declaration of all three charging pumps inoperable coincident with entry into abnormal operating procedures for a reactor coolant system leak due to the identification of a leaking pressurizer safety valve. The inspectors observed the operators' performance related to the assessment of operability, Technical Specification applicability, plant risk, procedures, and assessment of Emergency Action Level criteria.

- On June 22, 2015, operations following the loss of Isolated Phase Bus Duct Cooling, including a plant down power to 60%. The inspectors observed the operators performance related to abnormal operating procedure entry, plant power maneuvering and control, and the assessment of plant risk.

In addition, the inspectors assessed the operators' adherence to plant procedures, including the Shutdown Operations Protection Plan and other operations department policies.

These activities constitute completion of three quarterly licensed operator performance samples, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- April 27, 2015, shutdown cooling system review

- May 6, 2015, loss of control room air conditioning

- June 1, 2015, plant level performance criteria exceeded due to December 17, 2014, reactor plant trip

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

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

b. Findings

No Findings were identified.

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

a. Inspection Scope

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

- April 19, 2015, orange risk during spent fuel pool cooling pump discharge vent valve replacement

- April 22, 2015, yellow risk during reactor vessel head movement

- May 5, 2015, orange risk during component cooling water maintenance outage

- May 6, 2015, yellow risk during component cooling water maintenance outage

- May 14, 2015, yellow risk during shutdown operations at lowered reactor coolant system inventory and reduced time to boil

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

These activities constitute completion of five maintenance risk assessment inspection samples, as defined in Inspection Procedure 71111.13.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed five operability determinations and functionality assessments that the licensee performed for degraded or nonconforming structures, systems, or components (SSCs):

- April 17, 2015, operability assessment of high pressure safety injection pumps following a loss of cooling water

- June 9, 2015, operability assessment of steam generator auxiliary feedwater inlet valves for material suitability

- June 9, 2015, operability assessment of safety injection tank 6A check valves and relief valve leakage

- June 15, 2015, operability assessment of overstressed containment spray piping and supports inside containment

- June 19, 2015, operability assessment of switchgear room supplemental cooling

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

These activities constitute completion of five operability review samples, as defined in Inspection Procedure 71111.15.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

On May 18, 2015, the inspectors reviewed a permanent modification to the emergency diesel generator load shed scheme. The inspectors reviewed the design and implementation of the modification. The inspectors verified that work activities involved in implementing the modification did not adversely impact operator actions that may be required in response to an emergency or other unplanned event. The inspectors verified that the post-modification testing was adequate to establish the operability of the SSC as modified.

These activities constitute completion of one sample of permanent modifications, as defined in Inspection Procedure 71111.18.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- May 20, 2015, reactor coolant pump 3A post maintenance testing following seal replacement
- May 21, 2015, spent fuel pool vent valve AC-898 repair post maintenance test
- May 27, 2015, raw water pinhole leak repair post maintenance testing
- May 30, 2015, cable replacement on raw water pump C post maintenance testing
- June 7, 2015, steam generator auxiliary feedwater inlet valve HCV-1107A post maintenance testing
- June 24, 2015, instrument air valve regulator replacement for low pressure safety injection pump 1A discharge valve
- June 25, 2015, emergency diesel generator #1 relay replacement

The inspectors reviewed licensing and design basis documents for the SSCs and the maintenance and post-maintenance test procedures. The inspectors observed the performance of the post-maintenance tests and/or reviewed the test results to verify that

the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected SSCs.

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

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

During the station's refueling outage that concluded on June 10, 2015, the inspectors evaluated the licensee's outage activities. The inspectors verified that the licensee considered risk in developing and implementing the outage plan, appropriately managed personnel fatigue, and developed mitigation strategies for losses of key safety functions. This verification included the following:

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

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

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

Inservice tests:

- June 17, 2015, safety injection system category A and B valve exercise test

- June 18, 2015, raw water pump AC-10C quarterly inservice test

Other surveillance tests:

- May 18, 2015, recirculation actuation signal logic and switch test

- June 18, 2015, diesel driven auxiliary feedwater pump operability test

- June 29, 2015, channel A safety injection, containment spray and recirculation actuation signal test

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

These activities constitute completion of five surveillance test samples, as defined in Inspection Procedure 71111.22.

b. Findings

No Findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed an in-office review of,

- TBD-EPIP-OSC-1H, “Hazards and Other Conditions Affecting Plant Safety,” Revision 3, dated January 22, 2015;

- Radiological Emergency Response Plan, Section J, “Protective Response,” Revision 24, dated March 12, 2015; and

- EPIP EOF-7, “Protective Action Guidelines,” Revision 27, dated March 12, 2015.

These revisions,

- revised emergency action level HA1, “Natural or Destructive Phenomena affecting the protected area,” condition 6, to change the low river level entry condition to 975 feet, 8 inches;

- revised Emergency Plan Table J-2 and Procedure EOF-7, Attachment 1, “Protective Action Recommendations Based on Dose Assessment/Field Team Radiological Data,” by

    - removing consideration of releases of less than one hour duration (shelter between 1 rem TEDE and 2 rem TEDE, evacuate for 2 rem TEDE and above); and

- removing guidance about special groups who may require different protective action limits.

These revisions were compared to their previous revisions, to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, to Nuclear Energy Institute Report 99-01, “Emergency Action Level Methodology,” Revision 5, and to the standards in 10 CFR 50.47(b) to determine if the revisions adequately implemented the requirements of 10 CFR 50.54(q)(3) and 50.54(q)(4). The inspector verified that the revisions did not decrease the effectiveness of the emergency plan. These reviews were not documented in safety evaluation reports and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection.

These activities constitute completion of three emergency action level and emergency plan changes sample as defined in Inspection Procedure 71114.04.

b. Findings

Introduction. The NRC identified three examples of a Severity Level IV violation for the licensee’s failure to submit a summary of its analysis of the effect of changes to emergency plan implementing procedures on the site emergency plan in accordance with the requirements of 10 CFR 50.54(q)(5).

Description. The inspector reviewed Radiological Emergency Response Plan, Section J, “Protective Response,” Revision 24, TBD-EPIP-OSC-1H, “Hazards and Other Conditions Affecting Plant Safety,” Revision 3, and EPIP EOF-7, “Protective Action Guidelines,” Revision 27. The inspector noted that the cover page for each emergency plan change and each procedure change describes the reason for the change. The licensee did not include any information about the effect of the change on the site emergency plan. The inspector searched the Agencywide Document Management System (ADAMS) and did not locate any independent report by the licensee summarizing its analysis of the effect of the changes.

Analysis. The failure to submit a summary of the analysis of the effect of changes to emergency plan implementing procedures on the site emergency plan is a performance deficiency within the licensee’s ability to foresee and correct. The issue is more than minor because the licensee’s failure to submit the required summary affects the NRC’s ability to perform its regulatory function, and the licensee has not incorporated this requirement into its program. The inspectors evaluated the issue using Section 6.6.d of the NRC Enforcement Policy, dated July 12, 2011, and determined it to be a Severity Level IV violation because the issue involved the licensee’s ability to implement a regulatory requirement not related to assessment or notification. The issue was entered into the licensee’s corrective action program as Condition Report CR 2015-04934. Traditional enforcement violations are not assigned a cross-cutting aspect.

Enforcement. Title 10 of the Code of Federal Regulations, Part 50.54(q)(5), requires, in part, that a report of each change made after February 21, 2012, be made, including a summary of its [regulatory] analysis, within 30 days after the change is put into effect. Contrary to the above, the licensee did not report a summary of its analysis for TBD-EPIP-OSC-1H, “Hazards and Other Conditions Affecting Plant Safety,” Revision 3,

within 30 days after the changes were put into effect on January 22, 2015, and did not report a summary of its analysis for Radiological Emergency Response Plan, Section J, “Protective Response,” Revision 24, and EPIP EOF-7, “Protective Action Guidelines,” Revision 27, within 30 days after the changes were put into effect on March 12, 2015. The licensee did not report a summary of its analysis for these procedures between February 21 and June 18, 2015. Because this issue is of very low safety significance and the condition was entered into the licensee’s corrective action program as Condition Report 2015-04934, this violation is being treated as a non-cited violation in accordance with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2015002-008, Failure to Submit Summaries of the Impact of Changes to the Emergency Plan and Implementing Procedures.

2. RADIATION SAFETY

Cornerstones: Occupational and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors assessed the licensee’s performance in assessing the radiological hazards in the workplace associated with licensed activities. The inspectors assessed the licensee’s implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures. The inspectors walked down various portions of the plant and performed independent radiation dose rate measurements. The inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors reviewed licensee performance in the following areas:

- The hazard assessment program, including a review of the licensee’s evaluations of changes in plant operations and radiological surveys to detect dose rates, airborne radioactivity, and surface contamination levels

- Instructions and notices to workers, including labeling or marking containers of radioactive material, radiation work permits, actions for electronic dosimeter alarms, and changes to radiological conditions

- Programs and processes for control of sealed sources and release of potentially contaminated material from the radiologically controlled area, including survey performance, instrument sensitivity, release criteria, procedural guidance, and sealed source accountability

- Radiological hazards control and work coverage, including the adequacy of surveys, job coverage and contamination controls, the use of electronic dosimeters in high noise areas, dosimetry placement, airborne radioactivity monitoring, controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools, and posting and physical controls for high radiation areas and very high radiation areas

- Radiation worker and technician performance with respect to radiation protection work requirements

- Audits, self-assessments, and corrective action documents related to radiological hazard assessment and exposure controls since the last inspection

These activities constitute completion of one sample of radiological hazard assessment and exposure controls as defined in Inspection Procedure 71124.01.

b. Findings

No findings were identified.

2RS3 In-plant Airborne Radioactivity Control and Mitigation (71124.03)

a. Inspection Scope

The inspectors evaluated whether the licensee controlled in-plant airborne radioactivity concentrations were consistent with ALARA principles and that the use of respiratory protection devices did not pose an undue risk to the wearer. During the inspection, the inspectors interviewed licensee personnel, walked down various portions of the plant, and reviewed licensee performance in the following areas:

- The licensee's use, when applicable, of ventilation systems as part of its engineering controls

- The licensee's respiratory protection program for use, storage, maintenance, and quality assurance of NIOSH certified equipment, qualification and training of personnel, and user performance

- The licensee's capability for refilling and transporting SCBA air bottles to and from the control room, operations support center during emergency conditions, status of SCBA staged and ready for use in the plant, associated surveillance records, and personnel qualification and training

- Audits, self-assessments, corrective action documents related to in-plant airborne radioactivity control, and mitigation since the last inspection

These activities constitute completion of one sample of in-plant airborne radioactivity control and mitigation as defined in Inspection Procedure 71124.03.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical hours (IE01)

a. Inspection Scope

The inspectors reviewed licensee event reports for the period of April 1, 2014 through March 31, 2015 to determine the number of scrams that occurred. The inspectors compared the number of scrams reported to the number reported for the performance indicator. Additionally, the inspectors verified the number of critical hours during the period. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline, "Revision 7, to determine the accuracy of the data reported.

These activities constituted verification of the unplanned scrams per 7000 critical hours performance indicator as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Unplanned Power Changes per 7000 Critical Hours (IE03)

a. Inspection Scope

The inspectors reviewed operating logs, corrective action program records, inspection reports for the period of April 1, 2014, through March 31, 2015, to determine the number of unplanned power changes that occurred. The inspectors compared the number of unplanned power changes documented to the number reported for the performance indicator. Additionally, the inspectors verified the number of critical hours during the period. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the data reported.

These activities constituted verification of the unplanned power changes per 7000 critical hours performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.3 Unplanned Scrams with Complications (IE04)

a. Inspection Scope

The inspectors reviewed the licensee's basis for including or excluding in this performance indicator each scram that occurred between April 1, 2014, and March 31, 2015. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the data reported.

These activities constituted verification of the unplanned scrams with complications performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.4 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors verified there were no unplanned exposures or losses of radiological control over locked high radiation areas and very high radiation areas during the period of April 1, 2014, to March 31, 2015. The inspectors reviewed a sample of radiologically controlled area exit transactions showing exposures greater than 100 millirem. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

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

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed corrective action program records for liquid or gaseous effluent releases that occurred between April 1, 2014, and March 31, 2015, and were reported to the NRC to verify the performance indicator data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

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

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution

.1 Routine Review

a. Inspection Scope

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

problem identification and resolution activities during the performance of the other inspection activities documented in this report.

b. Findings

No findings were identified.

.2 Annual Follow-up of Selected Issues

a. Inspection Scope

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

- May 11, 2015, sand accumulation in the raw water system

  On May 11, 2015, the inspectors performed an in-depth review of the licensee's assessment and completed corrective actions associated with sand accumulation in the raw water system. On May 2, 2013, the licensee was performing preventative maintenance on containment air cooler backup raw water inlet valve HCV-400E and was unable to cycle the valve locally. The licensee performed troubleshooting and determined that sand had accumulated on the raw water side of the valve and that sand had prevented the valve from opening. Valve HCV-400E provides an alternate source of heat removal using the raw water system in the event that the component cooling water system is unavailable. This method of heat removal is not credited for any postulated design basis accident. The licensee performed an extent of condition review and determined that sand had also accumulated in piping that supplies other components that provide an alternate cooling path for the component cooling water system using the raw water system. The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews and compensatory actions. The inspectors verified that the licensee appropriately prioritized corrective actions and that these actions were adequate to remove the accumulated sand from the raw water system and to institute preventative measures to ensure that sand would not challenge operability of the raw water system.

- May 20, 2015, reactor cavity liner leakage and boric acid accumulation under the reactor vessel

  The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews and corrective actions from a condition report documenting reactor cavity liner leakage and boric acid accumulation under the reactor vessel in 2012. The inspectors identified that a corrective action to coat the cavity liner prior to filling the cavity for their refueling outage was cancelled. As a result, further leakage occurred during the 2015 refueling outage; however, the inspectors determined that the impact of this leakage was minor. The inspectors verified that the licensee appropriately evaluated the current condition for nuclear detector well cooling insulation and support functions, coating requirements, sump debris analysis, boric acid accumulation and evaluation, and license renewal commitments. The inspectors also verified

that corrective actions are in place to evaluate how to preclude this leakage during the next refueling outage.

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

b. Findings

No findings were identified.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors reviewed the licensee's corrective action program, performance indicators, system health reports, and other documentation to identify trends that might indicate the existence of a more significant safety issue. The inspectors verified that the licensee was taking corrective actions to address identified adverse trends.

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

b. Findings

No findings were identified.

Equipment Reliability

The inspectors identified equipment reliability at Fort Calhoun Station as a continuing adverse trend. During 2014, several challenges with equipment reliability and performance occurred that resulted in equipment unavailability, unplanned technical specification entries, and operator burdens. These issues often affected safety and technical specification systems and included component cooling water, raw water, chemical and volume control, and containment spray systems. In addition, non-safety related systems important to safe operation were affected and included switchgear room cooling, the diesel driven auxiliary feed water pump, and radiation monitors. Finally, inadvertent actuation of a sudden pressure relay on a unit auxiliary transformer resulted in a generator lockout, turbine trip, and subsequent reactor trip from 100% power. During the recent 2015 refueling outage, several equipment issues were identified that affected safety system availability and plant performance. Examples included a loss of all charging pumps, losses of shutdown cooling system capability, a reactor coolant system leak from a pressurizer relief valve, loss of control room ventilation, loss of auxiliary feed water to steam generators, and several boric acid leaks that complicated inspections of the reactor vessel head.

The inspectors discussed the outage issues with the licensee's senior leadership team prior to plant startup to determine how the licensee planned to assess the equipment reliability issues in aggregate. The licensee conducted a startup challenge review utilizing members from the operations, engineering, and outage organizations. The licensee reviewed the events that occurred during the outage and their corrective actions in response to each event. They also evaluated the safety of the plant with respect to a

decision to startup. The licensee identified that there were weak areas in knowledge of program requirements, understanding of system operation and response, inconsistent rigor and technical human performance, and failures to drive legacy items to resolution. The licensee also determined that startup was acceptable based on the correction of the individual issues, improvements driven by implementation of Exelon processes, and improvements to the facility equipment made during the outage. Also, several root- and apparent-cause investigations are in progress to determine causes and define long term corrective actions, including those intended to improve equipment reliability. This startup challenge review and evaluation is documented in CR 2015-07662. The inspectors determined that the licensee's assessment of the events that occurred during the outage and corrective actions were reasonable to continue with the startup.

In recognition of equipment reliability issues in general at the Fort Calhoun Station, the licensee's senior leadership team have implemented a number of engineering department excellence plans to restore engineering performance, address longterm development of engineering resources, partner with maintenance and operations departments in equipment reliability improvement activities, focus on the effective use of operating experience, and conduct self-assessments and corporate and industry benchmarking. Also, the licensee continues to utilize a Plant Health Committee Top 10 list to prioritize and focus the improvement of critical systems that have exhibited reliability and performance issues. The inspectors determined that these higher level actions should be effective in improving equipment reliability program performance at Fort Calhoun Station.

The inspectors will continue to monitor the licensee's progress in addressing and improving equipment reliability at the Fort Calhoun Station.

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

Plant Events

a. Inspection Scope

For the plant events listed below, the inspectors reviewed and observed plant parameters, reviewed personnel performance, and evaluated performance of mitigating systems as applicable. The inspectors communicated the plant events to appropriate regional personnel, and compared the event details with criteria contained in Inspection Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of potential reactive inspection activities. As applicable, the inspectors verified that the licensee made appropriate emergency classification assessments and properly reported the event in accordance with 10 CFR Parts 50.72 and 50.73. The inspectors reviewed the licensee's follow-up actions related to the event to assure that the licensee implemented appropriate corrective actions commensurate with their safety significance.

- Operator response to a loss of shutdown cooling pumps on May 29, 2015

- Operator response to an unplanned technical specification action statement entry when all three charging pumps were declared inoperable on June 1, 2015

- Operator response to a reactor coolant system leak when a pressurizer safety

valve started to leak by its seat on June 1, 2015

- Operator response to the failure of steam generator inlet auxiliary feedwater valves to operate normally in declaration of a loss of auxiliary feedwater on June 5, 2015, and issue follow up on June 7, 2015

- Operator response to the loss of isolated phase bus duct cooling and subsequent downpower to 60 percent power on June 22, 2015

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

b. Findings

Failure to Follow Instructions and Procedures Related to Snubber Activities

Introduction. The inspectors identified a non-cited violation of very low safety significance of 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures, and Drawings," because activities affecting quality were not accomplished in accordance with instructions and procedures established by the licensee. Specifically, the inspectors identified three examples where the licensee failed to follow site instructions and procedures associated with a safety related snubber that supported multiple trains of auxiliary feedwater (AFW).

Description. Bergen Paterson hydraulic snubbers are piping supports that allow slight movement of piping during normal operations, but restrict piping movement during seismic events and are utilized to ensure the operability of safety related equipment during such an event. These types of snubbers are safety related, and are required to be periodically tested and inspected to ensure their operability. Snubber FWS-5A is a safety related snubber that supports main feedwater piping to ensure the auxiliary feedwater alternate injection path remains operable during a seismic event.

On March 7, 2015, a Quality Control (QC) inspector was performing a visual inspection of snubber FWS-5A in accordance with licensee procedure QC-ST-HSS-0003, "Visual Inspection of Bergen Paterson Hydraulic Snubbers." The QC inspector identified that the upper spherical bearing on snubber FWS-5A was coming apart and as a result rejected the snubber. Following rejection, the QC inspector made a remark in the work package noting the degraded condition, but did not immediately inform his supervisor and the Shift Manager as required by procedure QC-ST-HSS-0003. In addition, the QC inspector did not document the degraded condition in a condition report as required by licensee instruction PI-AA-120, "Issue Identification and Screening Process."

On March 8, the day shift QC inspector discovered the snubber work package on his desk with a note describing the failed condition. The day shift QC inspector informed his supervisor and the Shift Manager of the degraded condition, and generated a condition report. He did not, however, report that the degraded condition had actually been discovered the previous day. The Shift Manager subsequently performed an Immediate Operability Determination, declared snubber FWS-5A inoperable, and entered Technical Specification (TS) Limiting Condition of Operation (LCO) 2.0.1(2)b. This LCO permits one snubber to be inoperable for a period up to 12 hours, provided that the licensee has assessed and managed the risk associated with the inoperable snubber. Upon the

expiration of the 12 hours, the LCO action statement requires that the licensee then declare the snubber's supported system inoperable and enter the applicable TS LCO for the supported system. After declaring snubber FWS-5A inoperable, the Shift Manager activated the Outage Control Center (OCC) for the emergent issue and notified the NRC resident inspectors of the unplanned TS LCO entry. The licensee removed the snubber from service to perform maintenance and repair or replace the snubber, however no parts were available to complete this action within the 12 hour timeframe so the nonconforming snubber was reinstalled. An engineering evaluation was subsequently issued approximately 30 minutes before the LCO action time expired concluding that the snubber would be able to perform its safety function and was therefore operable, but degraded, since the snubber bearing exhibited sufficient freedom of movement to account for design pipe stresses and movement.

The inspectors discussed the risk element of TS 2.0.1(2) with NRC Headquarters personnel and reviewed the Safety Evaluation Report (SER) associated with License Amendment No. 238, the license amendment that had implemented TS 2.0.1(2). A bounding risk assessment based on seismic risk contained within the SER provides the analysis for the acceptable 12 hour LCO timeframe with stipulations related to the implementation of the TS. Some of these stipulations include: (1) at least one auxiliary feedwater train not associated with the inoperable snubber, or some alternative means of core cooling [e.g. Power Operated Relief Valves (PORVs) and High Pressure Safety Injection (HPSI)] must be available; (2) the confirmation that at least one train of systems supported by the inoperable snubber would remain capable of performing their safety functions for nonseismic design loads; and (3) entry into the LCO must be done in accordance with an overall Configuration Risk Management Profile (CRMP) to ensure that potentially risk-significant configurations are identified and avoided. The licensee utilizes procedure SO-G-44, "Snubber List and Operability Requirements," to meet the stipulations of this SER, and this procedure identifies snubber FWS-5A as a safety related snubber that affects auxiliary feedwater injection flow and is applicable to the 12 hour timeframe. The inspectors identified that the licensee had not taken actions to ensure that PORVs and HPSI remained available, nor was entry into the LCO done in accordance with a CRMP to preclude a higher risk configuration, contrary to procedure SO-G-44.

Lastly, the inspectors reviewed licensee instruction OP-FC-108-104-FOL, "Technical Specification Limiting Conditions for Operation Action Requirements," and identified that the licensee did not utilize a time of discovery based on the initial recognition of the degraded condition on March 7, 2015, but instead utilized a time of discovery based on the time that the oncoming QC inspector identified the condition on March 8, 2015. This is significant since TS 2.0.1(2) is based on seismic risk and the bounding risk assessment associated with this TS as described in the SER is based on limiting the inoperable timeframe to 12 hours. Beyond 12 hours, the licensee would have been required to declare both trains of the auxiliary feedwater system inoperable.

The inspectors identified three examples where the licensee failed to follow licensee instructions and procedures related to snubber activities. Specifically:

(1) The QC inspector that identified the degraded condition on snubber FWS-5A did not report the condition to his immediate supervisor or the Shift Manager, contrary to procedure QC-HSS-0003. The NRC inspectors discussed this with the QC supervisor on March 9, 2015, and determined that no condition report

had been generated to document this procedural deficiency before that time.

(2) The licensee did not adequately perform a risk assessment or implement risk management actions associated with the inoperable snubber as described in the NRC's SER for TS LCO 2.0.1(2) and as proceduralized in procedure SO-G-44. These actions included ensuring that PORVs and HPSI remained available to remove decay heat since snubber FWS-5A affected both trains of auxiliary feedwater.

(3) The licensee utilized a time of discovery based on the oncoming QC inspector's discovery of the degraded condition on March 8. The actual discovery of the degraded condition had occurred on March 7, 2015, and although the control room was not aware of this when initially reported, the OCC became aware of this later in the day but did not inform the control room to adjust the time of discovery. The licensee did not utilize a time of discovery based upon the recognition of inoperability as described in procedure OP-FC-108-104-FOL.

The licensee performed an apparent cause analysis of the event and determined that station managers did not ensure that the necessary instructions and procedures related to snubber activities were well understood during the work planning process. As a result, the station did not adequately identify potential pitfalls related to snubber maintenance and inspections and did not adequately plan for these activities. The inspectors concluded that the licensee failed to implement a work process that ensured the identification and appropriate management of risk associated with snubber maintenance and inspections as required by station procedures and documented in the NRC SER for TS LCO 2.0.1(2).

<u>Analysis</u>. The inspectors determined that the licensee's failure to follow site instructions and procedures was a performance deficiency within the licensee's ability to foresee and correct and therefore should have been prevented. The finding is more than minor because if left uncorrected, the performance deficiency could have led to a more significant safety concern. Specifically, the failure to have a work process to ensure the appropriate identification and management of risk associated with safety related snubbers could result in unacceptable risk configurations that are not analyzed under technical specifications and could challenge the reliability of safety related equipment during a seismic event.

Using NRC Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2 "Mitigating System Screening Questions" Part B, dated July 1, 2012, the inspectors determined the finding to be of very low safety significance (Green) since the finding did not result in the loss of equipment specifically designed to mitigate a seismic initiating event as determined by a subsequent engineering evaluation. Specifically, the licensee concluded that the snubber would have been able to perform its function and was operable, but degraded, since the snubber bearing exhibited sufficient freedom of movement to account for design pipe movement.

The finding has a cross-cutting aspect in the area of Human Performance, the Work Management aspect, since the licensee did not implement a work process that ensured the identification and management of risk commensurate to the work (H.5).

Enforcement. Title 10 CFR Part 50 Appendix B, Criterion V requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, or a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Procedure QC-ST-HSS-0003, "Visual Inspection of Bergen Paterson Hydraulic Snubbers," required immediately informing supervision and the control room following a failed snubber inspection. Procedure SO-G-44, "Snubber List and Operability Requirements," required a risk assessment and risk management actions for an inoperable snubber. Procedure OP-FC-108-104-FOL, "Technical Specification Limiting Conditions for Operation Action Requirements," required a time of discovery based on recognition of snubber inoperability. Contrary to the above, between March 7 and March 8, 2015, activities affecting quality were not accomplished in accordance with instructions and procedures. Specifically, the licensee did not notify supervision and control room operators regarding a degraded condition associated with safety related seismic snubber, did not assess or manage the risk of the inoperable snubber, and did not use an appropriate time of discovery in accordance with licensee instructions and procedures. Immediate actions taken by the licensee to address this violation included a review of all other snubber inspections that were rejected to ensure that other degraded conditions were reported to the control room, a review of all planned snubber maintenance with respect to online risk, and the issuance of interim guidance to all Shift Managers on the subject of snubber operability and risk. Because this violation was of very low safety significance and was entered into the licensee's corrective action program as Condition Reports 2015-02808, 2015-02837, 2015-02857, and 2015-03073, this violation is being treated as an NCV, consistent with section 2.3.2 of the NRC's Enforcement Policy: NCV 05000285/2015002-008, "Failure to Follow Instructions and Procedures Related to Snubber Activities."

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000285/2015001-03, "Required Inservice Testing for Steam Generator Auxiliary Feedwater Inlet Valves."

The NRC opened this unresolved item because the inspectors identified that inservice testing performed on the steam generator auxiliary feedwater inlet valves was not consistent with the safety functions of the valves as described in design basis document EA11-026, "CQE Valves w/Essential Accumulators". EA11-026 identified that these valves have both an open and close active safety function, yet these valves were not being tested in the close direction as required by the applicable ASME OM Code for category B valves. The licensee's initial assessment of this discrepancy was that design basis document EA11-026 was incorrect, but more detailed analysis would be required to justify this assessment.

On May 26, 2015, the licensee completed a design review analysis of the steam generator AFW inlet valves and concluded that the valves do not have an active safety function to close, and that design basis document EA11-026 was incorrect. The licensee subsequently issued Condition Report 2015-7235 to address the discrepant statements in EA11-026.

The inspectors reviewed the licensee's design analysis of the steam generator auxiliary feedwater inlet valve and testing protocol and concluded that the licensee was meeting

the required inservice testing for these valves. In addition, since the licensee is tracking the revision to EA11-026 in their corrective action program, this URI is closed.

.2 (Closed) Unresolved Item (URI) 05000285/2013008-22, "Fort Calhoun Station's Ability to Classify Components as Safety-Related."

The NRC opened this unresolved item because an inspection team had noted that the NRC had identified several performance deficiencies prior to mid-2013 for not classifying structures, systems, and components as safety-related based on the safety function being performed. Subsequently, as described in its December 17, 2013, letter to the Omaha Public Power District (ML13351A395), the NRC confirmed that the licensee had committed to implement certain actions detailed in their December 2, 2013, letter to the NRC titled, "Integrated Report to Support Restart of Fort Calhoun Station and Post-Restart Commitments for Sustained Improvement," (ML13336A785). Some of those actions are associated with Design and Licensing Basis Control and Use, and one of those actions involves their ability to classify components as safety-related. Specifically, the licensee's letter stated, in part,

> "OPPD will complete a significant effort to perform a risk-focused reconstitution of the design basis, the licensing basis, and the Updated Safety Analysis Report. As part of this reconstitution, OPPD will ensure proper classification of equipment, convert to a safety related "Q List" approach for equipment classification and complete a key calculation review."

Because the licensee has committed to ensure proper classification of equipment through the subject reconstitution of the design basis, and because the NRC will inspect the results of that effort, this URI is closed to that inspection.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On May 8, 2015, the inspectors presented the radiation safety inspection results to Mr. L. Cortopassi, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On May 15, 2015, the inspectors presented the inservice inspection results to Mr. L. Cortopassi, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On June 18, 2015, the inspector conducted a telephonic exit meeting to present the results of the in-office inspection of changes to the licensee's emergency plan and emergency action levels to Mr. E. Plautz, Manager, Emergency Preparedness, and other members of the licensee staff. The licensee acknowledged the issues presented.

On July 24, 2015, the inspectors presented the integrated inspection results to Mr. L. Cortopassi and other members of the licensee's staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Anderson, Manager, Nuclear Projects
D. Bakalar, Manager, Site Security
D. Brehm, Supervisor, Radiation Protection
C. Cameron, Principle, Regulatory Specialist
L. Cortopassi, Site Vice President
E. Dean, Plant Manager
S. Fatora, Director, Site Work Management
H. Goodman, Director, Site Engineering
R. Hugenroth, Supervisor Nuclear Oversight
T. Hutchinson, Reliability Engineer
K. Ihnen, Manager, Nuclear Oversight
K. Kingston, Superintendent Maintenance
R. Lowery, Senior Operations Training Instructor
K. Maassen, Program Engineer
T. Maine, Manager, Radiation Protection
K. Mann, Regulatory Specialist
E. Matzke, Senior Regulatory Engineer
W. McCall, Health Physicist
J. McManis, Manager, Engineering Programs
B. Pearson, Supervisor, Radiological Protection
C. Scofield, Senior Nuclear Design Engineer-Mechanical
T. Simpkin, Manager, Site Regulatory Assurance
S. Swanson, Director, Site Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2015002-001	NCV	Failure to Include a Class 1 Component in the Reactor Vessel Pressure Boundary Integrity Test (Section 1R08.1)
05000285/2015002-002	NCV	Failure to Perform a Valid 40-Month Inservice Test (Section 1R08.1)
05000285/2015002-003	NCV	Failure to Establish Adequate Work Instructions to Clean and Inspect the Reactor Vessel Head (Section 1R08.2)
05000285/2015002-004	NCV	Failure to Incorporate Vendor Manual Recommendations for Conducting Preventative Maintenance on the Reactor Vessel Head Vent Valve (Section 1R08.3)
05000285/2015002-005	NCV	Failure to Promptly Identify and Correct a Condition Adverse to Quality Involving a Spent Fuel Pool Cooling Vent Valve Leak (Section 1R08.3)
05000285/2015002-006	NCV	Failure to Identify and Correct Loose Incore Instrument Nozzle Connection (Section 1R08.3)
05000285/2015002-007	FIN	Failure to Perform Functionality Assessments for the Spent Fuel Pool Cooling System (Section 1R08.5)
05000285/2015002-008	NCV	Failure to Submit Summaries of the Impact of Changes to the Emergency Plan and Implementing Procedures (Section 1EP4)
05000285/2015002-009	NCV	Failure to Follow Instructions and Procedures Related to Snubber Activities (Section 4OA3)

Closed

05000285/2015001-03	URI	Required Inservice Testing for Steam Generator Auxiliary Feedwater Inlet Valves (Section 4OA5)
05000285/2013008-22	URI	Fort Calhoun Station’s Ability to Classify Components as Safety-Related (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

Number	Title	Revision
AOP-01	Acts of Nature	42
OP-AA-108-111-D01	Severe Weather and Natural Disaster Guidelines	9
SOG-119	Site Wind Generated Missile Protection Standards	2

Section 1R04: Equipment Alignment

Procedures

Number	Title	Revision
OI-CC-0001	Component Cooling System Normal Operations	82
OI-ES-2	Engineered Safeguard Control, Cold Shutdown Operations	12
OI-RW-0001	Raw Water System Normal Operation	106
OI-RW-1	Normal Operation of Raw Water System	108
OI-SC-1	Shutdown Cooling Initiation	69
OI-SC-2	Shutdown Cooling Operation and Termination	31
OI-SC-3	Alternate Shutdown Cooling using Containment Spray Pumps	29
OI-SFP-1-CL-A	Spent Fuel Pool Cooling Startup Lineup	
OI-SFP-1	Spent Fuel Pool Cooling Normal Operation	39
OI-SFP-6	Spent Fuel Pool Heatup Rate	13
OP-AA-108-111D01	Severe Weather and Natural Disaster Guidelines	9
OP-12	Fueling Operations	71
OP-ST-RW-3003	Raw Water System Category A and B interface Valve Test	14
SO-O-21	Shutdown Operations Protection Plan	54

Miscellaneous Documents

Number	Title	Revision
DBD-SI-130	Shutdown Cooling Design Basis Document	23
USAR 9.3	Shutdown Cooling System	14
USAR 9.6	Spent Fuel Pool Cooling System	27

Condition Reports (CRs)

2014-10365	2015-7217	2015-7242	2015-7585

Drawings

Number	Title	Revision
2111	EM-115, Instrument Loop for Pressurizer Wide Range	18
12234	Simplified One Line Diagram Plant Electrical System P&ID	39
10441	Spent Fuel Pool Cooling Flow Diagram	0
14548	Shutdown Cooling System, Interconnection Diagram	7
36681	Shutdown Cooling System Flow Diagram	1

Section 1R05: Fire Protection

Procedures

Number	Title	Revision
SO-G-28	Station Fire Plan	87
SO-G-91	Control and Transportation of Combustible Material	30

Miscellaneous Documents

Number	Title	Revision
	AC-10D Raw Water Pump Maintenance Fire Risk Management Actions	
FC0581	UFHA Combustible Loading	0
FHA-EA-97-001	Fire hazards Analysis (FHA) Manual	17
USAR 9.11	Fire protection system	25

Drawings

Number	Title	Revision
B-4701	Sh. 1, Penetration Layout Drawing Fire Area 1	0

Section 1R06: Flood Protection Measures

Procedures

Number	Title	Revision
OI-RW-1	Raw Water System Normal Operation	108
	Flood Barrier Control Program	10

Condition Reports (CRs)

2015-04595	2015-07944

Miscellaneous Documents

Number	Title	Revision
	East Raw Water header Clearance Tag List	
FHA-EA-97-001	Fire hazards Analysis (FHA) Manual	17
EA08-10	Engineering Analysis Internal Flooding	0

Section 1R07: Heat Sink Performance

Procedures

Number	Title	Revision
SE-PRT-CCW-0002	Thermal Performance of Shutdown Cooling Heat Exchangers	7
PED-SEI-16	Evaluation of Heat Exchanger Performance	12

Work Orders

175542-01	175543-01

Miscellaneous Documents

Number	Title	Revision Date
	Preventive Maintenance Program Update	9/9/04
DBD-51-130	Shutdown Cooling Design Basis Document	23
LIC-89-045	Response to NRC Generic Letter 88-17	2/10/89
USAR-9.3	Shutdown Cooling System	14

Section 1R08: Inservice Inspection Activities

Procedures

Number	Title	Revision
SO-M-2	Standing Order	40
PED-SEI-46	Functional Equipment Group (FEG) and Functional Importance Determination (FID) Process	1
PED-SEI-13	Preventive Maintenance Program – Technical Basis	12
ER-AP-331	Boric Acid Corrosion Control (BACC) Program	5, 7
PBD-10	Boric Acid Corrosion Control Program	16,18
TD C631-0010	Clampseal Globe Valves Serving Instructions	1
ER-AA-200	Preventive Maintenance Program	1
OP-FC-108-115	Operability Determinations	2
PBD-2	Inservice Inspection Program	15
PI-AA-125	Corrective Action Program (CAP) Procedure	2
OP-ST-RC-3007	Periodic Reactor Coolant System Integrity Test	25
QC-ST-SFP-3001	Forty Month Inservice Test of the Spent Fuel Pool Cooling System	2
M1742P1	Inspection for Boric Acid Corrosion of Carbon and Low Alloy Steel Components	1
M1741	Visual Examination	3
M1741ALT	Alternative VT-2 Qualification	2
FCSG-68-6	Functional Importance Determination (FID) Process	1
OP-AA-108-108	Unit Restart Review	15
CY-AP-130-3100	Deposit Sampling and Analysis	1
PI-AA-120	Issue Identification and Screening Process	1
PED-GEI-55	Instructions for ASME Section XI Repair/Replacement Process	20
ER-AP-331-1005	Boric Acid Corrosion Control (BACC) Program Performance Indicators	4
ER-AP-331-1004	Boric Acid Corrosion Control (BACC) Training and Qualification	5
ER-AP-331-1003	RCS Leakage Monitoring and Action Plan	7
ER-AP-331-1002	Boric Acid Corrosion Control Program Identification, Screening, and Evaluation	8

Procedures

Number	Title	Revision
ER-AP-331-1001	Boric Acid Corrosion Control (BACC) Inspection, Locations, Implementation, and Inspection Guidelines	7
SG-SGMP-12-19	Steam Generator Operational Assessment for Fort Calhoun Station 11RFO	0
SG-SGMP-11-9	Fort Calhoun Station Steam Generator 11RFO Degradation Assessment Report	1
SG-CDME-08-20	Steam Generator Condition Monitoring and Operational Assessment for Fort Calhoun Station 08RFO	0
66216	Outage FCR27 Steam Generator Condition Monitoring and Operational Assessment	0

Condition Reports (CRs)

2015-05863	2015-05038	2015-03972	2015-02717	2015-05842
2015-02309	2015-04701	2015-04170	2015-03973	2015-04705
2015-04729	2015-05735	2015-05255	2015-04175	2015-04174
2015-04173	2015-04172	2015-04171	2014-15622	2013-17334
2013-15223	2009-00408	2011-09296	2009-02074	2011-03423
2015-05729	2015-05856	2015-05864	2015-05432	

Work Orders

551054	315718	471289	360542	331407
491564	310658	524500	424795	550470
199765	427170	551120	471096	390145
518874	369095	390145	331407	

Drawings

Number	Title	Revision
11405-M-11	Auxiliary Coolant Spent Fuel Pool Cooling System Flow Diagram	57
E-23866-210-110	Reactor Coolant System Flow Diagram	84

Section 1R11: Licensed Operator Requalification Program

Procedures

Number	Title	Revision
O1-ST-2	Turbine Generator Startup	34
O1-RC-2A	RCS Fill and Drain Operators	9
OP-AA-3	Reactivity Management	1
OP-2A	Plant Startup	120
RE-CPT-RX-0001	Post Refueling Core Physics Testing and Power Ascension	50
AOP-22	Abnormal Operating Procedure, Reactor Coolant Leak	0
AOP-27	Abnormal Operating Procedure, Generator Malfunctions	12

Miscellaneous Documents

Number	Title	Revision
	Emergency Action level Criteria Tables	0
TS 2.2.4	Charging Pumps-Operating	

Section 1R12: Maintenance Effectiveness

Procedures

Number	Title	Revision
ER-AA-310	Implementation of the maintenance rule	9
ER-AA-310-1001	Maintenance Rule – Scoping	4
ER-AA-310-1002	Maintenance Rule Functions – Safety Significance Classification	3
ER-AA-310-1003	Maintenance Rule – Performance Criteria Selection	4
ER-AA-310-1004	Maintenance Rule – Performance Monitoring	13
ER-AA-310-1005	Maintenance Rule – Dispositioning Between (a)(1) and (a)(2)	7
ER-AA-1006	Maintenance rule-expert panel roles and responsibilities	5

Miscellaneous Documents

Number	Title	Revision
RG 1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	3
NUMARC 93-01	Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	2

Condition Reports (CRs)

2009-0541	2011-7306	2012-1820	2013-02777
2013-8730	2014-14866	2015-01059	2015-04409

Work Orders

425200	425282	467735	475434

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

Number	Title	Revision
ER-AA-600	Risk Management	7
ER-AA-600-1011	Risk Management Program	13
FCSG-19	Performing Risk Assessments	17
MM-RR-RC-0305	Removal of Reactor Vessel Closure Head, Hold Down Ring and Upper Guide Structure	38
PE-ST-VX-3001	ASME Code Relief Valve Test for the CCW System	23
SO-G-96	Planned LCO Entry Criteria and Equipment Reliability Control	15
SO-G-123	Protected Equipment Program	8
SO-O-21	Orange Risk Contingency Plan, Attachment 6	54
SO-O-21	Shutdown Operations Protection Plan, Attachment 3	54
OI-CC-1	Component Cooling Water Operation	82
OI-RC-LA	RCS Fill and Drain Operations	90
OI-RW-1	Raw Water System Operation	14

Miscellaneous Documents

Number	Title	Revision Date
	Fort Calhoun Station 2015 Refueling Outage FCR27 Safety Plan	0
	Protected Equipment Scheme	5/5/15
OU-AA-103	Shutdown Safety Management Program	15
SDBD-AC-SFP-102	Spent Fuel Storage and Pool Cooling	24
USAR 9.6	Auxiliary Systems Spent Fuel Cooling System	10

Condition Reports (CRs)

2015-01059	2015-05598	2015-05600	2015-05652	2015-05733

Work Orders (WOs)

331407

Section 1R15: Operability Determinations and Functionality Assessments

Procedures

Number	Title	Revision
OP-AA-101-113-1004	Equipment Prompt	29

Condition Reports

2015-07633	2015-07638	2015-07716	2015-7583	2015-159
2013-02611	2013-1396	2014-12535	2015-04578	2015-04654
2015-06013	2015-05076			

Miscellaneous Documents

Number	Title	Revision
DBD-AFW-AFW117	Auxiliary Feedwater Design Basis Document	45
EA-FC-91-14	Effect of loss of cooling Water on SI/CS Pumps	9
EA-10-020	Electrical Equipment Qualification Radiation Dose Reconstitution Analysis	1
EA-13-040	Evaluation of Valves with Teflon Subcomponents Located in Radiation Areas	0
EC114970	SI/CS Pump Bearing Oil Upgrade	0
EC65926	Modify Containment Spray Header and Ring Pipe Support-Comp Measure to Support Operability	
FC08434	Operability Evaluation of Containment Spray Piping and Structural System	
USAR 6.2	Safety Injection System	0A
USAR 9.4	Auxiliary Feedwater System	22
	Equipment Failure Fault Tree Analysis	
USAR 15.3.5	Containment Liner Plate and Penetration Sleeve Fatigue	
50.59 Screening for EC65926 and EC64785	Screening for, Modify Containment Spray Header and Ring Pipe Supports Comp Measure to Support Operability 14-018	

Drawings

Number	Title	Revision
E-23866-210-130	Safety Injection and Containment Spray System Flow Diagram	66
E-23866-210-120	Chemical and Volume control system	26
E-23866-210-110	Reactor Coolant System Flow Diagram	16

Section 1R18: Plant Modifications

Procedures

Number	Title	Revision
FC-1134	Modification Construction Approval	12
HU-AA-1212	Attachment 5 Risk Rank and Augmented Review Guideline	4
LS-AA-104-1001	Review for EC58145	4
MA-AA-716-012	Post Maintenance Testing	19
OP-ST-ESF-0002	Diesel Generator No. 1 and No. 2 Auto Operation	65
OP-ST-ESF-0018	Engineered Safeguards Actuation Signal Retest	
PE-RR-VX-04145	Inspection and Repair of Safety Related Fisher "HSC" Control Valves	13
PED-GEI-3.10	MOD Reviewer Checklist and Independent Reviewer Checklist	13
PED-GEI-3.2	System Interaction Checklist	71
PED-GEI-4.1	Fire Protection System Interaction Checklist	12
PED-GEI-6.1	Pipe Rupture System Interaction Checklist	4
PED-GEI-8.0	Environmental Radiological Release Consideration Preliminary Screening Questions	3
PED-GEI-9.1	410 and 480Volt AC Power-Summary	16
SP-CP-08-480-1B4B	Calibration of the Protective Relays for 480-1B4B Bus	26

Condition Reports

2013-19342	2015-7564	2015-7565	2015-7580	2015-7583
2015-6725				

Miscellaneous Documents

Number	Title	Revision
	480 Volt Bus Drawings	
EC58145	Improved DG Load Shed Scheme	

Section 1R19: Post-Maintenance Testing

Procedures

Number	Title	Revision
IC-PM-FX-0600	Air/Nitrogen Regulator Replacement	22
IEEE 35549	Item Equivalency Evaluation for the Replacement of Low Friction Seal	
K-115325-PSW1-0001	Test procedure for Medium Voltage Cable Testing	0
MA-AA-013	Post Maintenance Testing Selection Instructions	22
MA-AA-716-012	Post Maintenance Testing	19
MA-AA-716-100	Maintenance Alterations Process	12
MM-RR-RC-001	Testing of Reactor Coolant Pump Seals	7
OP-FC-108-115	Operability Determinations	2
OP-ST-AFW-3010	Auxiliary Feedwater System Quarterly Category A and B Valve Exercise Test	12
OP-ST-DG-0001	Diesel Generator 1 Check	86
SO-M-103	System Cleanliness	15
SPP-7.1	Weld and Base metal repair	4
PE-RR-VX-04145	Inspection and Repair of Safety Related Fisher “HSC” Control Valves	13
WPS-801	Welding Procedure Specification	14

Condition Reports (CRs)

2012-16746	2013-00273	2015-03972	2013-19342	2015-03972
2015-6130	2015-7564	2015-7565	2015-7580	2015-7583
2015-08097	2015-08098	2015-08113	2015-08163	2015-08205
2015-08199				

Work Orders (WOs)

382108	452767	491564-01	524711-05	540618-05
550466-01	550470-01	552468-02		

Miscellaneous Documents

Number	Title	Revision Date
EC 27564	Update of Reactor Coolant Pump Mechanical Seals	0

Number	Title	Revision Date
EC45641	AC-898 SFP 5A and 5B Discharge Header Vent Valve Replacement	0
EC63269	Replace Cable E68 for AC-10C-M Raw Water Pump Motor	0
EC66130	Fluorcarbon (FKM) O-Rings in HCV-1107A/B and HCV-1108A/B	
IEE383-1974	IEEE Standard for Type Test of Class 1E Electrical Cables, Field Splices, and Connections for Nuclear Power Generating Stations	4/30/1975
K-115325-PSW1-0001	Test Procedure for Medium Voltage cable Testing	0
PED-GE1-7	Specification of Post-Modification Test Criteria	15a
SDBD-RC-128	Reactor Coolant System Design Basis Document	35
SPP-7.1	Weld and Base Metal Repair	4
TD B580 0020	Instruction for Reactor Coolant Pump	1
USAR 9.4	Auxiliary Feedwater System	22
USAR 9.6	Spent Fuel Pool Cooling System	10
USAR 9.8	Raw Water System	35
DBD-AC-RW-101	Raw Water Design Basis Document	39
USA B31.7	US Standard Code for Pressure Piping	8/24/1966

Drawings

Number	Title	Revision
1C-3070	Seal Cartridge Test Final Assembly	2
2C-4829	Cooling Water Piping for Reactor Coolant Pump	
2F-1165	DFSS Vertical Recirculating Pump, Reactor Coolant	
11405-M-11	Auxiliary Coolant, Spent Fuel Cooling System	57
D-5210	Raw Water Pumps Conduit Details	0
B-4250	Cable Block Diagram AC-1C	1

Section 1R20: Refueling and Other Outages

Procedures

Number	Title	Revision Date
AOP-13	Loss of Control Room Air Conditioning	10
AOP-17	Loss of Instrument Air	15
AOP-26	Turbine Malfunctions	10
AOP-32	Loss of 480V Busses	20
FC08443	Spent Fuel Pool Heat Up Rate for Refueling Outage 27	0
FCR27	Outage Safety Plan	12
FC95A	Shutdown Shift Turnover Log	10
HU-AA-1211	Pre-Job Briefings	0
HU-AA-1211-F-03	OP-3A Plant Shutdown, IPA Briefing Worksheet	0
HU-AA-1211-F-03	RE-CPT-RX-0001, Post Refueling Core Physics Testing and Power Ascension	0
HV-AA-1211	IPA Briefings	0
LIC-07-0005	Licensee Event report 2006-2008	1/26/07
	2015 Refueling Outage Safety Plan	4/12/15
LIC-88-1106	Response to NRC Generic Letter 88-17	1/4/88
MM-RR-RC-0314	Installation of Reactor Vessel Head	0
NF-AA-309	Attachment 9, PWR Move Sheet Package Verification Checklist for RF027, 4/1/15	27
OI-AFW-4	Auxiliary Feedwater Startup and System Operation	89
OI-CH-1	Chemical and Volume Control System Normal Operation	93
OI-CH-3	Chemical and Volume Control System Normal Operation of Volume Control Tank	
OI-CO-1	Containment Closeout	47
OI-EE-2	480V AC System Normal Operation	125
OI-FW-5	Steam Generator Blowdown Normal Operation	35
OI-FW-6	Steam Generator Draining	46
OI-RC-2A	RCS Fill and Drain Operations	90
OI-RC-4A	Pressurizer Cooldown and Venting	27

Number	Title	Revision Date
OI-RC-9	Reactor Coolant Pump Operation	77
OI-RPS-1	Reactor Protection System	12
OI-SC-5	Shutdown Cooling Purification	32
OI-SFP-1	Spent Fuel Pool Cooling Normal Operating	39
OI-SFP-6	Spent Fuel Pool Heatup Rate	13
OP-1	Master Checklist for Plant Startup	118
OP-2A	Plant Startup	120
OP-3A	Plant Shutdown	86
OP-12	Fueling Operations	71
OP-ST-ESF-0006	Engineered Safety Features Off-Site Power Low Signal (OPLS) Functional Test	33
OP-ST-ESF-0011	Channel A and B Automatic and Manual Engineered Safeguard Actuation Signal Test	43
OP-ST-ESF-0019	Recirculation Actuation Signal Logic and Switch Test	24
OU-AA-103	Shutdown Safety Contingency Plan	15
SO-G-92	Conduct of Infrequently Performed Procedures	17
SO-G-107	Storage of Transient Equipment and Material to Prevent Seismic Interactions or Tornado Pressurization	10
SO-O-21	Shutdown Operations Protection Plan	54
SO-G-23	Surveillance Test Program	65
SO-G-107	Storage of Transport Material	10
TDB-11	Cycle 27 Boron Concentration	34
TDB-III.1.a	Temperature Correction for Pressurizer Level Indicators LI-101X/Y	6
TDB-III.2	Actual level in Pressurizer vs. Indicated Level in Pressurizer	7
TDB-III.3.a	Temperature Correction for Pressurizer Level Indicators LI-101X/Y	6
TDB-111.42	Requirements for ECCS and Containment Equipment Operation in Mode 3, Transition between Modes	
TDB-III.7.d	RCS Pressure and Temperature Limits	8
TDB-III.8	RCS Mass vs. RCS Temperature	3

Number	Title	Revision Date
TDB-III.42	Requirements for ECCS and Containment Cooling Equipment Operation in Mode 3, Transition between Modes 3 and 4 and Mode 4 and 5	
TDB-IV.10.a	Acid Reducing Conditions	0
TDB-IV.10b	Acid Reducing Conditions	0
TDB-VIII	Equipment Applicability Guidance	64

Condition Reports (CRs)

2012-05067	2013-15223	2014-15357		
2015-3838	2015-0410	2015-04037	2015-04057	2015-04069
2015-04380	2015-02808	2015-04442	2015-04460	2015-04466
2015-04474	2015-04509	2015-04511	2015-04610	2015-04624
2015-04701	2015-04729	2015-04777	2015-05163	2015-05255
2015-05268	2015-05270	20145-05271	2015-05302	2015-05304
2015-05329	2015-05336	2015-05384	2015-05432	2015-05439
2015-05524	2015-05631	2015-05735	2015-05817	2015-05836
2015-05852	2015-05856	2015-05858	2015-05861	2015-05862
2015-05864	2015-05870	2015-05930	2015-06013	2015-06063
2015-06130	2015-06158	2015-06300	2015-06557	2015-06577
2015-06611	2015-06618	2015-06663	2015-06725	2015-06735
2015-06743	2015-06758	2015-06761	2015-06770	2015-06774
2015-06831	2015-06878	2015-06890	2015-06902	2015-06959
2015-07198	2015-07324	2015-07344	2015-07345	2015-07426
2015-07429	2015-07467	2015-07479	2015-07487	2015-07716

Miscellaneous Documents

Number	Title	Revision
	Spent Fuel Pool Cooling Safety Function Decision Tree Basis Document	0

Work Orders

331407-01 471096-02

471096-02

Section 1R22: Surveillance Testing

Procedures

Number	Title	Revision
IC-ST-SA-3001B	Diesel Generator #2 Starting Air Compressors Discharge Check Valves Exercise Test	4
OI-VA-2	Attachment 11, Supplementary Cooling and Temperature Limits for Auxiliary Building Spaces	45
OP-FC-102-106	Operator Response Time Program	3
OP-PM-AFW-0004	Third Auxiliary Feedwater Pump Operability Verification	22
OP-ST-ESF-0009	Channel A Safety Injection, Containment Spray and Recirculation Actuation Signal Test	59a
OP-ST-ESF-0019	Recirculation Actuation Signal Logic and Switch Test	24
OP-ST-RW-3021	AC-10C Raw Water Pump Quarterly Inservice Test	40
OP-ST-SI-3001	Safety Injection System Category A and B Valve Exercise Test	37a

Miscellaneous Documents

Number	Title	Revision
TS 3.1	Instrumentation and Control, Table 3-2, Minimum Frequencies for Checks, Calibrations and Testing of Engineered Safety Features, Instrumentation, and Controls	
TS 3.3	Reactor Coolant System and Other Components Subject to ASME XI Boiler and Pressure Vessel Code Inspection and Testing Surveillance	
TS 2.4	Containment Cooling	
TDB 111.33	AC-10C Pump Curve	29
USAR 7.3	Engineering Safeguards Controls and Instrumentation	
	50.59 Evaluation for OI-VA-2	
	50.59 Evaluation for EC-65749	
EC65749	Increasing Portable Fan Size and Changing the Flow Path for Switchgear Room Supplemental Cooling	0
FC06102	Switchgear Room Heatup Rate and Supplemental Flow Requirements after a Loss of Normal HVAC at 100% Reactor Power	4

Number	Title	Revision
FC07889	HELB Environmental Analysis for FCS Auxiliary Building, Room 81	0

Condition Reports (CRs)

2014-0072	2014-1460	2014-11223	2015-04191	2015-6475
2015-08218	2015-08224	2015-08310	2015-08311	2015-08312
2015-08313				

Section 2RS1: Radiological Hazard Assessment and Exposure Controls Section

Procedures

Number	Title	Revision
RP-AA-10	Radiation Protection Process Description	3
RP-AA-16	ALARA Program Description	0
RP-AA-18	Radiological Posting and Labeling Program Description	1
RP-AA-19	High Radiation Area Program Description	2
RP-AA-100	Conduct of Radiation Protection Operations	0
RP-AA-210	Dosimetry Issue, Usage, and Control	25
RP-AA-300	Radiological Survey Program	12
RP-AA-300-1005	Removing Items from the SFP, Reactor Cavity, and Equipment Pit	1
RP-AA-301	Radiological Air Sampling Program	8
RP-AA-376	Radiological Postings, Labeling, and Markings	8
RP-AA-400	ALARA Program	11
RP-AA-410	Selection, Use, and Control of Protective Clothing	6
RP-AA-460	Control for High and Locked High Radiation Areas	26
RP-AA-800	Control, Inventory, and Leak Testing of Radioactive Sources	7
RP-FC-1	Radiation Protection	0
RP-215	Refueling Shutdown, Forced Outages, and Plant Start-Up Initial Actions and Radiological Survey Procedure	14
RP-229	Changes in Radiological Conditions Due to Plant Evolutions	2

Number	Title	Revision
RP-ST-RM-0002	Radioactive Material Source Surveillance	12

Audits, Self-Assessments, and Surveillances

Number	Title	Date
PI-AA-126-1001	2014 RP/Chemistry Technical Support Self-Assessment	9/5/14

Radiological Surveys

Number	Title	Date
M-20150423-12	Cavity Survey for Tracking UGSLR Bushings	4/23/15
M-20150420-04	CTMT 1036’	4/20/15
M-20150419-26	Blind Flange Dose Verification Survey	4/19/15
M-20150429-15	Room 62 CVCS Ion Exchanger Room	4/28/15
M-201400529-1	Room 62 CVCS Ion Exchanger Room	5/29/14
M-20150428-13	A S/G Bowl Survey	4/28/15
M-20150428-9	A S/G Bowl/Manways	4/28/15
M-20150428-11	B S/G Bowl	4/28/15
M-201500428-12	B S/G Bowl/Manways	4/28/15
M-20141211-1	Room 509	12/10/14
M-20150305-1	Room 509	3/4/15
M-20150420-1	Containment Elevation 994	4/20/15
M-20150412-44	Containment Elevation 994	4/12/15

Condition Reports (CRs)

2015-02863	2015-02552	2015-00833	2015-04319	2014-12740
2014-10291	2014-10093	2014-09410	2014-09371	2014-07197
2014-07828	2014-09928			

Radiation Work Permits

Number	Title	Revision
15-0503	Aux Building Insulation	0
15-0604	CTMT Outage Decon Activities with Added Controls	0
15-0613	Rx Head Disassembly/Reassembly	0
15-0614	Rx Head/Upper Internals Moves	0
15-0619	Nuclear Instrumentation and Undervessel	0
15-0641	Transfer Cart Wheels Measurements	0
15-0643	PZR Heater Replacement	0

Number	Title	Revision
15-0651	Boric Acid Cleaning	0
15-0702	S/G RP Support	0

Miscellaneous Documents

Number	Title	Date
FC-RP-ST-RM-2	Radioactive Source Inventory and Leak Test	5/15/14
FC-RP-ST-RM-2	Radioactive Source Inventory and Leak Test	11/13/14
FC-1217	Non-Fuel Material Spent Fuel Pool Inventory Ledger	4/14/15

Section 2RS3: In-plant Airborne Radioactivity Control and Mitigation

Procedures

Number	Title	Revision
RP-AA-870-1001	Set-Up and Operation of Portable Air Filtration Equipment	3
RP-507	Inspection and Maintenance of Respiratory Protection Equipment	24
RP-510	Operation of Respirator Cleaning Equipment	9
RP-511	Recharging of SCBA Cylinders	9
RP-513	SCBA Air Compressor Fill System Operation	17
RP-AA-13	Respirator Protection Program	0
RP-AD-500	Respirator Protection Program	19
RP-AA-440	Respirator Protection Program	10
RW-700	HEPA Ventilation and HEPA Vacuum Program	8
RW-706	Leak Testing of HEPA Filtered Vacuum Cleaners and/or HEPA Ventilation Units	3
RP-FC-825-AD-513	SCBA Air Compressor Fill System Operation	0
RP-FC-440-AD-502	Use of Respirator Protection Equipment	0
OI-VI-1	Containment Heating, Cooling and Ventilation Systems Normal Operation	84
OI-VI-2	Auxiliary Building Ventilation System Normal Operation	45

Audits, Self-Assessments, and Surveillances

Number	Title	Date
RA-2014-1801-004	Respiratory Protection Self-Assessment 2015	1/31/15
2013-3618	2014 RP-Chem Technical Support Self-Assessment	9/5/14
NOSA-FCS-13-58	Radiation Protection Audit Report	7/19/13

Condition Reports (CRs)

2015-01220	2015-01218	2015-01223	2015-02677	2014-11085
2015-01123	2015-01019	2015-01045	2015-01031	2015-00948

Miscellaneous Documents

Number	Title	Date
SE-ST-VA-015	VA-64A Control Rm. Charcoal Filter & Methyl Iodide Test	2/6/15
SE-ST-VA-008	VA-64A Control Rm. Charcoal Filter & Methyl Iodide Test	2/3/15
SE-ST-VA-007	VA-64A Control Room Charcoal & HEPA Filter Test	1/24/15
SE-ST-VA-006	VA-64A Control Room Charcoal & HEPA Filter Test	2/5/15

Section 4OA1: Performance Indicator Verification

Miscellaneous

Number	Title	Revision Date
	Operating Logs	
LER 2014-007	Licensee Event Report, Plant Trip Due to Moisture Intrusion in a Transformer Control Cabinet	0
	Quarterly Baseline Inspection Reports from the 2nd Quarter 2014 through the 1st Quarter 2015	
	Operating Data Reports	
	NRC Performance Indicator Data Submittals	
	NRC Performance Indicators	

Section 4OA2: Problem Identification and Resolution

Condition Reports (CRs)

2012-05067	2012-00116	2013-09861	2013-12619	2013-17191
2013-23218	2015-06766	2013-04408	2013-09861	2015-06554
2015-06559	2015-5930	2014-06557	2015-6895	2015-06769
2015-06739				

Miscellaneous

Number	Title	Revision Date
DBD-101	Raw Water Design Basis	39
USAR 9.8	Raw Water System	35
EE66361	Evaluation of Reactor Vessel Leakage and Boric Acid Accumulation	0
FC02522	Sand and Sediment in CCW and W Systems Assessment	11/5/13

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

Procedures

Number	Title	Revision
OP-FC-108-111	Adverse Condition Monitoring and Contingency Planning	1
AOP-27	Abnormal Operating Procedure, Generator Malfunctions	12

Condition Reports (CRs)

2015-7293	2015-7328	2015-7330	2015-08116	2015-08124

Miscellaneous

Number	Title	Revision Date
	Emergency Action Level Criteria Tables, Figure 8.1.1	
	Abnormal Operating Procedure 22, Reactor Coolant Leak	35
TS 2.2.4	Charging Pumps-Operating	
TS2.1.1	Reactor Coolant System	
	Timeline of Isophase Bus Duct Cooling Fans Tripping	
	Adverse Condition Monitoring Plan, Isophase Bus Duct Cooling MCC Ambient Condition Monitoring	6/23/2015

Section 4OA5: Other Activities

Condition Reports (CRs)

2013-19107	2013-20057	2013-21519	2013-8095	2006-3381
2013-4733	2013-15336	2013-3669	2014-13158	

Work Orders (WOs)

502802-01	502803-01	502804-01	502804-03