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

October 2, 2012

Louis P. Cortopassi, Site Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

Subject: FORT CALHOUN - NRC INTEGRATED INSPECTION REPORT NUMBER 05000285/2012004

Dear Mr. Cortopassi:

On August 18, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station Unit 1. The enclosed inspection report documents the inspection results which were discussed on August 27, 2012, with you and other members of your staff.

The inspection(s) examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. The inspections were performed by resident and regional inspectors focusing on daily station activities and progress being made addressing items associated with the Station Restart Checklist enclosed in the Confirmatory Action Letter dated June 11, 2012.

Two NRC identified findings of very low safety significance (Green) were identified during this inspection. Both of these findings were determined to involve violations of NRC requirements. Additionally, the NRC has determined that one traditional enforcement Severity Level IV non-cited violation occurred.

Additionally, one violation of NRC requirements was identified. This finding was determined to be a violation related to a previously issued Red finding regarding circumstances surrounding the fire that resulted in a loss of power to six of nine safety-related 480 Vac buses and the resulting declaration of an Alert which occurred on June 7, 2011 (Inspection Reports 05000285/2011014 and 05000285/2012010; ML12072A128 and ML12101A193, respectively). The significance of this finding was bounded by the Red finding and therefore was not characterized by color significance. This finding was determined to involve a violation of NRC requirements. A separate citation will not be issued as this item is being evaluated by the NRC under the Manual Chapter 0350, "Oversight of Reactor Facilities in a Shutdown Condition Due to Significant Performance and/or Operational Concerns," process.

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

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC Resident Inspector at Fort Calhoun Station.

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

Sincerely,

/RA/

Michael C. Hay, Chief
Project Branch F
Division of Reactor Projects

Docket No.: 50-285
License No.: DPR-40

Enclosures: NRC Inspection Report 05000285/2012004
w/Attachment: Supplemental Information

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ADAMS: ☐ No ☑ Yes	☑ SUNSI Review Complete	Reviewer Initials: MCH
	☑ Publicly Available	☑ Non-Sensitive
	☐ Non-publicly Available	☐ Sensitive

SRI:DRP/F	RI:DRP/F	SPE:DRP/F	C:DRS/PSB2	ROE:DRP/F	SPE:DRP/F
JCKirkland	JFWingebach	RWDeese	JDRAKE	AKlett	ARosebrook
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2012004
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	July 1 through August 18, 2012
Inspectors:	J. Kirkland, Senior Resident Inspector J. Wingebach, Resident Inspector B. Tharakan, Resident Inspector A. Klett, Reactor Operations Engineer A. Rosebrook, Senior Project Engineer R. Deese, Senior Project Engineer F. Ramirez, Resident Inspector K. Clayton, Senior Operations Engineer C. Young, Senior Resident Inspector L. Carson II, Senior Health Physicist N. Greene, Ph.D., Health Physicist C. Alldredge, Health Physicist J. O'Donnell, Health Physicist
Approved By:	Michael C. Hay, Chief, Project Branch F Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2012004; 07/01/2012 – 08/18/2012; Fort Calhoun Station, Integrated Resident and Regional Report; Radioactive Gaseous and Liquid Effluent Treatment; Radiological Environmental Monitoring Program; and Radioactive Solid Waste Processing, and Radioactive Material Handling, Storage, and Transportation.

The report covered a 6-week period of inspection by resident and regional inspectors focusing on daily station activities and progress being made addressing items associated with the Station Restart Checklist enclosed in the Confirmatory Action Letter dated June 11, 2012. Additionally, an announced baseline inspection by region-based inspectors was performed. Three violations of low significance were identified; two Green non-cited violations, and one Severity Level IV non-cited violation. Additionally, one violation was identified, and was determined to be a violation related to and bounded by a previously issued Red finding regarding circumstances surrounding the fire that resulted in a loss of power to six of nine safety-related 480 Vac buses and the resulting declaration of an Alert which occurred on June 7, 2011, and therefore was not characterized by color significance. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The cross-cutting aspect is determined using Inspection Manual Chapter 0310, "Components Within the Cross Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. <u>NRC-Identified Findings and Self-Revealing Findings</u>

Cornerstone: Initiating Events

- <u>N/A</u>: The team identified a violation of 10 CFR 50 Appendix B Criteria III, "Design Control." Specifically, the design modification package for the 480 VAC breaker replacements failed to ensure the breaker coordination for the 480 VAC electrical buses was maintained. As a result, feeder breaker 1B3A tripped unexpectedly during the fire event in the 1B4A switchgear. This performance deficiency also resulted in the loss of multiple buses on both trains of 480 VAC, including ECCS systems, from a single fault on a 480 VAC bus. This finding and its corrective actions will be managed by the NRC's Inspection Manual Chapter 0350 Oversight Panel. This finding is associated with Enforcement Action 12-121.

  The failure to ensure that the 480 VAC electrical power distribution system design requirements were maintained was a performance deficiency that was within OPPD's ability to foresee and prevent. The performance deficiency was reviewed using NRC Inspection Manual Chapter 0612, Appendix B, "Issue Screening," and the issue was determined to be more than minor because it affected the Initiating Events Cornerstone attributes of protection against external events (i.e., fire) and design control. The issue adversely affected the associated cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The significance of this finding is bounded by the significance of the Red finding documented in Inspection Report 05000285/2012010. The licensee entered this issue into its corrective action program as CR 2011-6621. The performance deficiency had a cross-cutting aspect in the area of human performance associated with resources because OPPD failed to ensure that station procedures for engineering changes, plant modifications, inspections, installations, and maintenance contained sufficient details [H.2(c)] (Section 40A4).

Cornerstone: Miscellaneous

- SLIV. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73(a) for the failure to submit a Licensee Event Report within 60 days after the discovery of performing an operation prohibited by technical specifications. The licensee failed to report to the NRC that they moved fuel while the Spent Fuel Pool Area Charcoal Filtration System, VA-66, was not in operation, contrary to Technical Specification 2.8.3(4). The licensee discovered in September 2011 that the fuel movement in December 2009 was inappropriate based on technical specifications, but failed to submit Licensee Event Report, 2012-008-0 until July 27, 2012. This issue was entered into the licensee's corrective action program and evaluated with an Apparent Cause Analysis under Condition Report, 2012-08521 and 2012-08386

  The failure to make an official report to the NRC regarding an operation prohibited by the Technical Specifications is a performance deficiency. The issue was dispositioned using traditional enforcement because failing to submit the Licensee Event Report had the potential to adversely impact the NRC's ability to perform its regulatory function. The issue is characterized as a Severity Level IV violation in accordance with the NRC Enforcement Policy, Section 6.9.d.9. Since this issue was dispositioned using traditional enforcement, there is no cross-cutting aspect (Section 2RS06).

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of very low safety significance of Technical Specification 2.8.3(4), the limiting condition for refueling operations in the spent fuel pool. In December 2009, the licensee performed refueling operations with the Spent Fuel Pool Area Charcoal Filtration System, VA-66, declared inoperable. The failure to establish an operable Spent Fuel Pool Area Charcoal Filtration System, VA-66, before moving spent fuel was a performance deficiency and a violation of Technical Specification 2.8.3(4). The licensee entered this issue into the corrective action program as Condition Reports 2012-08521, 2012-0836 and Licensee Event Report 2012-008-0.

  The performance deficiency was determined to be more than minor because it adversely impacted the attribute of the Barrier Integrity Cornerstone objective to maintain radiological filtration functionality during operations in the spent fuel pool to protect the public from radionuclide releases caused by accidents or events. Using IMC 0609 Appendix A, “Barrier Integrity Significance Determination Process,” the inspectors determined this finding to be of very low safety significance (Green). Although fuel movements were contrary to the licensee’s technical specifications limiting condition for refueling operations, the finding represented a degradation of the radiological barrier function provided for the spent fuel pool fuel building. This finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not effectively incorporate internal operating experience and lessons learned from previous VA-66 ventilation system failures during spent fuel pool refueling operations and plant safety. Specifically, the licensee failed to systematically collect, evaluate, and communicate to affected internal stakeholders in a timely manner relevant internal and external operating experience, [P2(a)] (Section 2RS06).

Cornerstone: Public Radiation Cornerstone

- Green. Inspectors identified two examples of a non-cited violation of very low safety significance of Technical Specification 5.8.1 for the failure to adequately establish, implement, and maintain procedures for: (1) the onsite meteorological monitoring systems; and (2) reporting meteorological data in accordance with the Offsite Dose Calculation Manual requirements. The licensee entered these issues into the corrective action program as Condition Reports 2012-05658, 2012-05724 and 2012-05777.

  The failure to establish, implement, and maintain procedures to ensure the meteorological monitoring equipment is operable and required meteorological data is reported was a performance deficiency. This finding is more than minor because it affected the Public Radiation Safety cornerstone attribute of program and process. The failure to have and use applicable procedures to ensure the operability of the meteorological monitoring system and the accuracy of the Annual Radiological Effluent Release Report has the potential to impair public dose assessments of routine and accidental radioactive effluent releases. Using IMC 0609 Appendix D, “Public Radiation Safety Significance Determination Process,” the inspectors determined this finding to be of very low safety significance because the finding did not represent a significant degradation of the ability to assess dose to members of the public and the actual releases were well below established limits for members of the public. This finding has a cross-cutting aspect in the human performance area associated with the resources component because the licensee failed to ensure that personnel, procedures, and other resources were adequate for the operability of the meteorological monitoring system and implementation of Offsite Dose Calculation Manual requirements related to the annual effluent report, [H.2(c)] (Section 2RS07).

B. <u>Licensee-Identified Violations</u>

None.

REPORT DETAILS

Summary of Plant Status

The station remained shutdown in Mode 5 for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the Raw Water / Component Cooling Water Heat Exchanger AC-1D The inspectors verified that performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, “Heat Exchanger Performance Monitoring Guidelines”; the licensee properly utilized biofouling controls; the licensee’s heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchanger was correctly categorized under 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one heat sink inspection sample as defined in Inspection Procedure 71111.07-05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS06 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

a. Inspection Scope

This area was inspected to: (1) ensure the gaseous and liquid effluent processing systems are maintained so radiological discharges are properly mitigated, monitored, and evaluated with respect to public exposure; (2) ensure abnormal radioactive gaseous or liquid discharges and conditions, when effluent radiation monitors are out-of-service, are controlled in accordance with the applicable regulatory requirements and licensee procedures; (3) verify the licensee’s quality control program ensures the radioactive effluent sampling and analysis requirements are satisfied so discharges of radioactive materials are adequately quantified and evaluated; and (4) verify the adequacy of public dose projections resulting from radioactive effluent discharges. The inspectors used the requirements in 10 CFR Part 20; 10 CFR Part 50, Appendices A and I; 40 CFR Part 190; the Offsite Dose Calculation Manual, and licensee procedures required by the Technical Specifications as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed and/or observed the following items:

- Radiological effluent release reports since the previous inspection and reports related to the effluent program issued since the previous inspection

- Effluent program implementing procedures, including sampling, monitor setpoint determinations and dose calculations

- Equipment configuration and flow paths of selected gaseous and liquid discharge system components, filtered ventilation system material condition, and significant changes to their effluent release points, if any, and associated 10 CFR 50.59 reviews

- Selected portions of the routine processing and discharge of radioactive gaseous and liquid effluents (including sample collection and analysis)

- Controls used to ensure representative sampling and appropriate compensatory sampling

- Results of the inter-laboratory comparison program

- Effluent stack flow rates

- Surveillance test results of technical specification required ventilation effluent discharge systems since the previous inspection

- Significant changes in reported dose values

- A selection of radioactive liquid and gaseous waste discharge permits

- Part 61 analyses and methods used to determine which isotopes are included in the source term

- Offsite dose calculation manual changes

- Meteorological dispersion and deposition factors

- Latest land use census

- Records of any abnormal gaseous or liquid tank discharges

- Groundwater monitoring results

- Changes to the licensee's written program for indentifying and controlling contaminated spills or leaks to groundwater, if any

- Identified leakage or spill events and entries made into 10 CFR 50.75 (g) records, if any, and associated evaluations of the extent of the contamination and the radiological source term

- Offsite notifications~~,~~ and reports of events associated with spills, leaks, or groundwater monitoring results, if any

- Audits, self-assessments, reports, and corrective action documents related to radioactive gaseous and liquid effluent treatment systems since the last inspection

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

These activities constitute completion of the one required sample, as defined in Inspection Procedure 71124.06-05.

b. Findings

(1) Introduction. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73(a) for the failure to submit a Licensee Event Report within 60 days after the discovery of performing an operation prohibited by technical specifications. The licensee failed to report to the NRC that they moved fuel while the Spent Fuel Pool Area Charcoal Filtration System, VA-66, was not in operation, contrary to Technical Specification 2.8.3(4).

Description. The spent fuel storage decontamination area's air treatment system is designed to filter the building atmosphere flowing to the auxiliary building vent during refueling operations. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. In-place testing is performed to confirm the integrity of the filtration system. Technical Specification 3.2, Table 3.5, Section 10b requires, in part, that within 31 days after removal, a laboratory test of a sample of the charcoal adsorber show methyliodide penetration of less than 10 percent. Records reviewed showed that system VA-66 failed to meet this acceptance criterion three times within the last six surveillance test periods. This observation was noted in the licensee's Apparent Cause Analysis on the recurring surveillance test failures.

Technical Specification 2.8.3(4) restricts movement of fuel when the Spent Fuel Pool Area Ventilation system is inoperable because any movement of fuel when this system cannot perform its intended function creates the possibility of releasing radionuclides to the atmosphere without proper filtration. On November 16, 2009, the Spent Fuel Pool Area Charcoal Filtration System (VA-66) was sampled and failed to meet its required surveillance test acceptance criteria. The results of the surveillance test, SE-ST-VA-0010, measured an actual penetration of 11.52 percent and the VA-66 system was declared inoperable. On December 1, 2009, the licensee moved fuel during a core reload while VA-66 was still declared inoperable.

As a result of performing an operation prohibited by the Technical Specifications, the licensee was required by 10 CFR 50.73(a)(2)(i)(b), “Licensee Event Report,” to report the event to the NRC within 60 days after the discovery. The licensee determined on September 28, 2011, that, contrary to TS 2.8.3(4), fuel movement was conducted with the Spent Fuel Pool Area Ventilation system inoperable.

After the inspectors’ questions regarding the licensee’s failure to submit a timely LER to the NRC, they submitted Licensee Event Report 2012-008-0 on July 27, 2012. The licensee indicated that fuel movement was conducted while the Spent Fuel Pool Area charcoal filter was in service, yet not able to meet the adsorption criteria, hence inoperable. The licensee stated that they had previously concluded the event was not reportable because VA-66 was no longer credited in the fuel handling accident analysis. The inspectors noted that the licensee failed to recognize that the technical specification non-compliance required them to report the event.

This issue was entered into the licensee’s corrective action program and evaluated under Condition Reports 2011-07800, 2012-08521, 2012-08386, and Licensee Event Report 2012-008-0.

Analysis. The failure to make an official report to the NRC regarding an operation prohibited by the Technical Specifications is a performance deficiency. The issue was dispositioned using traditional enforcement because failing to submit the Licensee Event Report had the potential to impact the NRC’s ability to perform its regulatory function. The issue is characterized as a Severity Level IV violation in accordance with the NRC Enforcement Policy, Section 6.9.d.9 Since this issue was dispositioned using traditional enforcement, there is no cross-cutting aspect.

Enforcement. 10 CFR 50.73(a)(1) requires, in part, that the licensee shall submit a Licensee Event Report for any event of the type described in this paragraph within 60 days after the discovery of the event. 10 CFR 50.73(a)(2)(i)(B) requires, in part, that the licensee shall report any operation or condition which was prohibited by the plant’s Technical Specifications. Technical Specification 2.8.3(4) requires, in part, that with the spent fuel pool area ventilation system not in operation, the licensee shall suspend refueling operations in the spent fuel pool. Contrary to the above, in September 2011, the licensee discovered that in December 2009 they moved fuel in the spent fuel pool without the spent fuel pool area ventilation system in operation and failed to submit a Licensee Event Report within 60 days of discovery.

This violation is being treated as a noncited violation in accordance with the NRC Enforcement Policy, Section 2.3.2(a)(1): NCV 05000285/2012004-01, “Failure to report an event to the NRC within 60 days for an operation prohibited by Technical Specifications."

(2) Introduction. In December 2009, the licensee performed refueling operations with the Spent Fuel Pool Area Charcoal Filtration System, VA-66, declared inoperable. The failure to have an operable Spent Fuel Pool Area Charcoal Filtration System, VA-66, prior to moving spent fuel was a violation of very low safety significance associated with Technical Specification 2.8.3(4) limiting condition for refueling operations.

Description. Technical Specification 2.8.3(4) states that the spent fuel pool area ventilation system contains a charcoal filter to prevent release of significant radionuclides to the outside atmosphere. This spent fuel pool area ventilation system must be aligned and started before using the spent fuel pool during refueling operations. When the spent fuel pool area ventilation system is not in operation the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended. This effectively mitigates a radiological release during a postulated fuel handling accident scenario.

As discussed in LER 2012-008-0, submitted on July 27, 2012, the charcoal adsorber in VA-66 was not operable and fuel movement occurred in December of 2009 because the licensee failed to recognize they were not in compliance with Technical Specification 2.8.3(4) limiting condition for refueling operations. This issue was entered into the licensee's corrective action program and evaluated under Condition Report 2011-07800, 2012-08521, 2012-08386 and Licensee Event Reports 2012-008-0.

Analysis. The performance deficiency was determined to be more than minor because it adversely impacted the attribute of the Barrier Integrity Cornerstone objective to maintain radiological functionality to protect the public from radionuclide releases caused by accidents or events. Using IMC 0609 Appendix A, "Barrier Integrity Significance Determination Process," the inspectors determined this finding to be of very low safety significance (Green). Although fuel movements were contrary to the licensee's technical specifications limiting condition for refueling operations, the finding did not represent a loss of the radiological barrier function provided for the spent fuel pool fuel building. This finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not effectively incorporate internal operating experience and lessons learned from previous VA-66 ventilation system failures during spent fuel pool refueling operations. Specifically, the licensee failed to systematically collect, evaluate, and communicate to affected internal stakeholders in a timely manner relevant internal and external operating experience, [P.2(a)].

Enforcement. The Limiting Condition for Refueling Operation in the spent fuel pool Technical Specification 2.8.3(4) states, in part, that the spent fuel pool area ventilation system shall be in operation. With the spent fuel pool area ventilation not in operation, suspend refueling operations in the spent fuel pool. Contrary to the above, on December 1, 2009, the licensee performed refueling operations and moved spent fuel without the spent fuel pool area ventilation system in operation. This violation of very low safety significance was entered into the licensee's corrective action program. This violation is being treated as a noncited violation in accordance with the NRC Enforcement Policy, Section 2.3.2(a). NCV 05000285/2012004-02, "Fuel Move with SFP Ventilation Inoperable a Condition Prohibited by Technical Specification 2.8.3(4)."

2RS07 Radiological Environmental Monitoring Program (71124.07)

a. Inspection Scope

This area was inspected to: (1) ensure that the radiological environmental monitoring program verifies the impact of radioactive effluent releases to the environment and sufficiently validates the integrity of the radioactive gaseous and liquid effluent release program; (2) verify that the radiological environmental monitoring program is implemented consistent with the licensee's technical specifications and/or offsite dose calculation manual, and to validate that the radioactive effluent release program meets the design objective contained in Appendix I to 10 CFR Part 50; and (3) ensure that the radiological environmental monitoring program monitors non-effluent exposure pathways, is based on sound principles and assumptions, and validates that doses to members of the public are within the dose limits of 10 CFR Part 20 and 40 CFR Part 190, as applicable. The inspectors reviewed and/or observed the following items:

- Annual environmental monitoring reports and offsite dose calculation manual
- Selected air sampling and thermoluminescence dosimeter monitoring stations
- Collection and preparation of environmental samples
- Operability, calibration, and maintenance of meteorological instruments
- Selected events documented in the annual environmental monitoring report which involved a missed sample, inoperable sampler, lost thermoluminescence dosimeter, or anomalous measurement
- Selected structures, systems, or components that may contain licensed material and has a credible mechanism for licensed material to reach ground water
- Records required by 10 CFR 50.75(g)
- Significant changes made by the licensee to the offsite dose calculation manual as the result of changes to the land census or sampler station modifications since the last inspection

- Calibration and maintenance records for selected air samplers, composite water samplers, and environmental sample radiation measurement instrumentation

- Interlaboratory comparison program results

- Audits, self-assessments, reports, and corrective action documents related to the radiological environmental monitoring program since the last inspection

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

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.07-05.

b. Findings

Introduction. The inspectors identified two examples of a green non-cited violation of very low safety significance of Technical Specification 5.8.1 for the failure to adequately establish, implement, and maintain procedures for: (1) the onsite meteorological monitoring systems and (2) reporting meteorological data in accordance with the Offsite Dose Calculation Manual (ODCM) requirements. Specifically, procedures did not exist to maintain the operability of the meteorological monitoring station and procedures did not exist to ensure the Annual Radiological Effluent Release Report contained the meteorological information required by the ODCM.

Description. No procedures existed to ensure operability of the meteorological tower and the availability of onsite meteorological data. Licensee Chemistry procedure CH-AD-0049, "Annual Meteorological Data," states, in part, that this procedure provides a means to ensure onsite tower operation. However, no written instructions existed to ensure meteorological tower monitoring operations were implemented that also ensured the availability of onsite meteorological data. In addition, the licensee failed to establish and implement adequate procedures for programs specified in Technical Specification 5.16. Specifically, the licensee did not have written procedures to ensure ODCM required meteorological data was included in the Annual Radiological Effluent Release Report.

Licensee procedure CH-ODCM-0001, "Off-Site Dose Calculation Manual," Section 5.2.1 states, in part, that "The Radioactive Effluent Release Report shall include a summary of the meteorological conditions concurrent with the release of airborne effluents during each quarter as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants,' June 1974, Revision 1." Section F, "Meteorological Data," of this regulatory guide states that the Radioactive Effluent Release Report should include a cumulative joint frequency distribution for quarterly periods, similar data reported separately for meteorological conditions during batch releases, and tables, similar to Table 4A, separately for each stability class and elevation. Licensee implementing procedure CH-AD-0050, "Annual Radioactive Effluent Release Report," failed to adequately address and implement the reporting requirements in the ODCM and Regulatory Guide 1.21. Specifically, the 2010 and 2011 Annual Radiological Effluent Release Reports failed to include the quarterly meteorological conditions for gaseous effluent releases, similar tables of meteorological conditions for each measurement elevation, and the meteorological data for batch releases. The licensee initiated corrective actions to address the ODCM requirements for meteorological data in future annual effluent reports.

These issues were entered into the corrective action program as Condition Reports 2012-05658, 2012-05724, and 2012-05777.

Analysis. The failure to establish, implement, and maintain procedures to ensure meteorological monitoring is operable and required meteorological data is reported is a performance deficiency. This finding is more than minor because it affected the Public Radiation Safety cornerstone attribute of program and process, in that the failure to have and use applicable procedures to ensure the operability of the meteorological monitoring system and the accuracy of the Annual Radiological Effluent Release Report has the potential to impair public dose assessments of routine radioactive effluent releases. Using IMC 0609 Appendix D, "Public Radiation Safety Significance Determination Process," the inspectors determined this finding to be of very low safety significance because the finding did not represent a significant degradation of the ability to assess dose to members of the public and the actual releases were well below established limits for members of the public. This finding has a cross-cutting aspect in the human performance area associated with the resources component because the licensee failed to ensure that personnel, procedures, and other resources were adequate for ensuring the operability of the meteorological monitoring system and the implementation of ODCM requirements relating to the annual effluent report. [H.2(c)]

Enforcement. Technical Specification (TS) 5.8.1 requires, in part, that written procedures and policies shall be established, implemented, and maintained covering: (a) the applicable procedures recommended Appendix A of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, February 1978," and (d) all programs specified in Technical Specification 5.11 through 5.21, including TS 5.16, "Radiological Effluents and Environmental Monitoring Programs." Contrary to the above, as of June 21, 2012, the licensee failed to establish, implement, and maintain procedures in the two examples stated below.

(1) Meteorological monitoring procedures were not established for maintaining the operability of the onsite meteorological monitoring station per Section 7.h of Regulatory Guide 1.33 when the site was flooded from June 2011 to March 2012. (TS 5.8.1.a)

(2) The ODCM and subordinate procedures were not established or adequately implemented to ensure quarterly summaries of site meteorological data were included in the Annual Radiological Effluent Release Reports. Specifically, the 2010 and 2011 Annual Radiological Effluent Release Reports did not

contain the meteorological information required by the ODCM, Section 5.2.1 and Regulatory Guide 1.21, Revision 1. (TS 5.8.1.d)

Since this violation is of very low safety significance and was entered into the corrective action program as Condition Reports 2012-5658, 2012-5724, and 2012-5777, this violation is being treated as a noncited violation, consistent with Section 2.3.2(a) of the NRC Enforcement Policy: NCV 05000285/2012004-03; "Failure to Establish and Implement Adequate Procedures for Meteorological Monitoring and the Off-Site Dose Calculation Manual."

2RS08 Radioactive Solid Waste Processing, and Radioactive Material Handling, Storage, and Transportation (71124.08)

a. Inspection Scope

This area was inspected to verify the effectiveness of the licensee's programs for processing, handling, storage, and transportation of radioactive material. The inspectors used the requirements of 10 CFR Parts 20, 61, and 71 and Department of Transportation regulations contained in 49 CFR Parts 171-180 for determining compliance. The inspectors interviewed licensee personnel and reviewed the following items:

- The solid radioactive waste system description, process control program, and the scope of the licensee's audit program

- Control of radioactive waste storage areas including container labeling/marking and monitoring containers for deformation or signs of waste decomposition

- Changes to the liquid and solid waste processing system configuration including a review of waste processing equipment that is not operational or abandoned in place

- Radio-chemical sample analysis results for radioactive waste streams and use of scaling factors and calculations to account for difficult-to-measure radionuclides

- Processes for waste classification including use of scaling factors and 10 CFR Part 61 analysis

- Shipment packaging, surveying, labeling, marking, placarding, vehicle checking, driver instructing, and preparation of the disposal manifest

- Audits, self-assessments, reports, and corrective action reports radioactive solid waste processing, and radioactive material handling, storage, and transportation performed since the last inspection

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

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.08-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

.1 (Open) Licensee Event Report 05000285/2012-001-00: Inadequate Flooding Protection Procedure

During a review of the station's procedures for responding to external flooding conditions, it was determined that the guidance is not adequate to mitigate a design basis flood event (1014 feet mean sea level (msl)).

A root cause analysis is in progress. Following completion of the cause analysis a revision to this LER will be submitted to provide the results of the analysis.

Compensatory actions have been identified and are being implemented. Additional corrective actions are being evaluated by the licensee.

.2 (Open) Licensee Event Report 05000285/2012-002-00: Inadequate Qualifications for Containment Penetrations Renders Containment Inoperable

During a review of environmental qualification records for reactor containment building electrical penetrations, six penetrations were identified that may not provide an adequate seal during worst case (Design Basis Accident (DBA)) conditions as required. These penetrations are through wall from the containment into the auxiliary building.

A cause analysis is in progress and the results will be included in a supplement to this LER.

The station is currently in a refueling mode. Corrective actions to address the causes of this condition will be documented in the supplement to this LER.The subject penetrations will be restored to full environmental qualifications prior to plant startup.

.3 (Open) Licensee Event Report 05000285/2012-003-00: Non-Conservative Error in Calculation for Alternate Hot Leg Injection Results in Unanalyzed Condition

A non-conservative error was identified in the input calculation for post-LOCA cooling flow (post-RAS (recirculation actuation signal)). The calculation used an incorrect (non-conservative) input for LPSI pump performance. The associated procedure (EOP/AOP Attachment 11) as written does not provide adequate direction during the Alternate Hot Leg Injection mode of operation. Therefore, the procedural guidance may not ensure the completion of the safety function of providing adequate core cooling during the Alternate Hot Leg Injection mode of operation under a worst case scenario.

A cause analysis is in progress and the results will be included in a supplement to this LER.

Corrective actions to address the causes of this condition will be documented in a supplement to this LER.

.4 (Open and Closed) Licensee Event Report 05000285/2012-004-00: Inadequate Analysis of Drift Affects Safety Related Equipment

While investigating operating experience from another station concerning potential instrument drift it was determined that Fort Calhoun Station (FCS) is subject to similar conditions. It was determined that pressure switches that provide safety related signals for high containment pressure to the reactor protection system (RPS) and engineered safeguards actuation circuitry may be similarly affected at FCS. The impact of the potential drift was evaluated and it was determined that neither RPS nor the engineered safeguard circuitry may actuate at the required containment pressure of 5 psig. An evaluation determined that the actuation may not occur until slightly higher than the required pressure. Other systems are currently being evaluated for this condition.

A cause analysis is being performed and will be provided in a supplement to this report.

Corrective actions will be determined following the completion of the cause analysis.

The licensee event report is closed. Revision 1 of this licensee event report was submitted on August 10, 2012.

.5 (Open) Licensee Event Report 05000285/2012-004-01: Inadequate Analysis of Drift Affects Safety Related Equipment

While investigating operating experience from another station concerning potential instrument drift it was determined that Fort Calhoun Station (FCS) is subject to similar conditions. It was determined that pressure switches that provide safety related signals for high containment pressure to the reactor protection system (RPS) and engineered safeguards actuation circuitry may be similarly affected at FCS. The impact of the potential drift was evaluated and it was determined that neither RPS nor the engineered safeguard circuitry may actuate at the required containment pressure of 5 psig. An evaluation determined that the actuation may not occur until slightly higher than the required pressure. Other systems are currently being evaluated for this condition.

A cause analysis was completed. However, internal reviews have identified that additional investigation is required to sufficiently characterize this issue. The results of the revised cause analysis and corrective actions will be published in a supplement to this report.

.6 <u>(Open and Closed) Licensee Event Report 05000285/2012-005-00: TS violation due to inadequate testing of DG fuel pumps</u>

During a QA review of surveillance procedures, an apparent failure to perform surveillance testing of the full automatic functions of the fuel oil transfer pumps was identified. Technical Specifications require monthly testing of the fuel oil transfer pumps. However, procedure changes made in 1990 removed the test of the automatic start of the fuel oil transfer pumps on low level in the Emergency Diesel Generator (EDG) day tank. Without full testing of the automatic functions of the fuel oil transfer pumps, they cannot be considered operable. Consequently, the EDGs cannot be considered operable because all auxiliary equipment to support operability has not demonstrated that it is fully capable of performing its safety function. There is reasonable assurance that the EDGs and fuel transfer pumps would function as required as the low level switches are calibrated on a refueling frequency and have functioned correctly during extended EDG surveillances. This report is being made in accordance with 10 CFR 50.73(a)(2)(i)(B), 50.73(a)(2)(ii)(B), 50.73(a)(2)(vii), and 50.73(a)(2)(v)(B) and (D). Corrective actions have been developed to revise the Emergency Diesel Generator surveillances to include fuel oil transfer pump surveillance testing.

The licensee event report is closed. Revision 1 of this licensee event report was submitted on June 1, 2012.

.7 <u>(Open) Licensee Event Report 05000285/2012-005-01: TS violation due to inadequate testing of DG fuel pumps</u>

On February 21, 2012, during a review of Fort Calhoun Station surveillance procedures, it was identified that the Emergency Diesel Generator (EDG) fuel oil transfer pumps have not been tested in accordance with the requirements of Technical Specifications (TSs). The inadequate testing was caused by a procedure change made in 1990 that removed the required monthly test of the automatic low level start feature of the fuel oil transfer pumps. There is reasonable assurance that the EDGs and fuel transfer pumps would function as required as the low level switches are calibrated on a refueling frequency.

The apparent cause of this event is a lack of technical rigor in the procedure change process employed in the1990’s. Corrective actions have been developed to revise the EDG surveillances to include fuel oil transfer pump surveillance testing.

.8 <u>(Open) Licensee Event Report 05000285/2012-006-00: Operation of Component Cooling Pumps Outside of the Manufacturers Recommendation</u>

The station identified that the CCW pumps were operating beyond their pump curves with the motor running into the service factor, runout conditions were not observed as there were no fluctuations in pressure, no fluctuations in motor amps, no visible signs of pitting or damage on impeller vain trailing, no damage to internal pump casing surfaces, no abnormal vibration, and no abnormal noise. A review of this condition determined that operation in this condition is a violation of plant technical specification for CCW operation.

A cause analysis is in progress. The results of the analysis will be published in a supplement to this LER.

.9 <u>(Open) Licensee Event Report 05000285/2012-007-00: Failure of Pressurizer Heater Sheath</u>

During inspections to determine the physical integrity of a failed pressurizer heater it was determined that the heater sheath (number 26) was cracked. Due to the location of the pressurizer heater crack, this is considered a degradation of the reactor coolant system boundary. The initial visual inspection of heater 26 in November 2011 did not identify the cracking. During efforts to remove the heater, a crack was observed on May 19, 2012. The crack is above and below the heater support plate. The crack is an axial crack showing some branching. The crack is about an inch above and inch below the heater support plate. These inspections were being performed as a result of operating experience. On May 23, 2012, it was determined that the pressurizer heater sheath was part of the reactor coolant system boundary.

A root cause analysis is in progress. The results will be published in a supplement to this LER.

The heater sheath has been removed and replaced. The other heater sheaths have been inspected and none of them had indications of cracking.

.10 <u>(Open) Licensee Event Report 05000285/2012-008-00: Technical Specification Violation for Fuel Movement (VA-66)</u>

A review of previously completed cause analyses has identified that Fort Calhoun Station (FCS) has moved fuel while the Spent Fuel Pool Area ventilation charcoal filter (VA-66) was inoperable due to failing the methyliodide penetration surveillance. FCS Technical Specification 2.8.3(4) requires the Spent Fuel Pool Area ventilation system to be in service prior to fuel movement. The Spent Fuel Pool Area ventilation system includes a charcoal filter which prevents the release of radioactive material to the outside atmosphere in the event of a fuel handling accident. However, the fuel handling accident analysis does not credit removal of any radioiodine through operation of the Spent Fuel Pool charcoal filter (VA-66); offsite radiological consequences are well within the 10 CFR 50.67 requirements without the charcoal filtration. There have been repeated charcoal efficiency test failures since 2005. There was evidence that the charcoal filters were not capable of meeting the 18-month surveillance frequency. Fuel movement was conducted while the Spent Fuel Pool Area charcoal filter was in service, yet potentially not able to meet the adsorption criteria, hence inoperable which is a violation of TS requirements.

A cause analysis is in progress. The results will be published in a supplement to this LER. Corrective actions included a revision of the applicable procedure to ensure that charcoal life is predicted and charcoal filter change out is performed before the charcoal expires.

.11 <u>(Open) Licensee Event Report 05000285/2012-009-00: Inoperable Equipment due to Lack of Environmental Qualifications</u>

During the review of the current analysis of record for Main Steam Line Break (MSLB) inside containment, no analysis or evaluation could be found to address why the original Electrical Environmental Qualification (EEQ) evaluation of peak MSLB conditions remain valid. The current analysis of record establishes that containment temperatures remain above the Loss of Coolant Accident (LOCA) peak temperature for substantially longer (220 seconds versus 60 seconds) but at a lower temperature (347.9 degrees Fahrenheit vs. 401 degrees Fahrenheit). The longer dwell times could result in a more adverse impact on environmentally qualified equipment.

A cause analysis is being processed and the results will be reported in a supplement to this LER.

Fort Calhoun Station will perform thermal lag analyses for the Electrical Equipment Qualification Program equipment located within containment prior to plant startup. The LER will be supplemented with the information from the EEQ and cause analysis.

.12 <u>(Open) Licensee Event Report 05000285/2012-010-00: Seismic Qualification of Instrument Racks</u>

While preparing an engineering package to relocate two transmitters, Fort Calhoun Station engineering identified seismic class 1 components in a seismic class 2 instrument rack. The instrument racks in the auxiliary building and containment were assessed with respect to the Updated Safety Analysis Report (USAR) specified class 1 requirements for seismic design. The seismic calculations for two instrument racks were over the analyzed weight for the seismic analysis. The instruments on these racks are used for reactor coolant system (RCS) pressure transmitters. During a seismic event, the excessive weight of these instrument racks could cause the racks to fail, resulting in an unisolable leak from the RCS. A cause analysis is in progress. The results of the analysis will be published in a supplement to this LER.

.13 <u>(Open) Licensee Event Report 05000285/2012-011-00: Emergency Diesel Inoperability Due to Bus Loads During a LOOP</u>

An Engineering review identified that a potential issue existed concerning Emergency Diesel Generators (EDG) capability to power required loads in certain loss of offsite power (LOOP) scenarios, specifically those scenarios during which a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB) does not occur. In a LOOP without a concurrent accident signal, the 480 V load shed that would be initiated as a direct result of the accident signal does not occur. Therefore, the electrical load that the EDGs must pick up when the EDG output breaker automatically closes could be significantly higher than the dead load that exists in an accident scenario. A review of design basis calculations and engineering analyses has identified several evaluations that consider the EDG dead load during accidents. However, no documents evaluating EDG dead loads in non-accident conditions were found. If one EDG were inoperable due to maintenance or other activities and the electrical distribution system loading conditions were such that the other EDG could have reached the output breaker trip settings during a LOOP event, both EDGs would be inoperable and FCS would have to take action per Technical Specification (TS) 2.0.1. It is conservative to assume that such conditions existed for those EDG outages that exceeded six hours. However, actions were not taken for two inoperable EDGs per the requirements of TS 2.0.1, resulting in operation or condition prohibited by TS.

A cause analysis is in progress. The results of the analysis will be published in a supplement to this LER.

.14 <u>(Open) Licensee Event Report 05000285/2012-012-00: Multiple Safety Injection Tanks Rendered Inoperable</u>

Fort Calhoun Station (FCS) operating procedures allow filling and sluicing multiple safety injection tanks (SITs) while at power, rendering the SITs inoperable during the evolution. The use of this procedure allowed multiple safety injection tanks to be concurrently filled while FCS was at power. FCS Technical Specifications (TS) and accident analysis do not allow more than one SIT to be inoperable. This condition was identified on March 19, 2012, while the unit was in Mode 5, by the NRC during initial license examination preparation.

A cause analysis is in progress. The results of the analysis will be published in a supplement to this LER.

.15 <u>(Open) Licensee Event Report 05000285/2012-013-00: Inadequate Calculation of Uncertainty Results a Technical Specification Violation</u>

Technical Data Book Procedure (TDB)-III.40, “Technical Specification Required SIRWT Levels,” lists the administrative requirements to maintain the Technical Specification (TS) required Safety Injection Refueling Water tank (SIRWT) levels. The required SIRWT level for TS 2.3 accounts for instrument uncertainty, as described in the basis for TS 2.3. However, the required SIRWT levels listed in TDB-III.40 for TS 2.2.7 and 2.2.8 do not account for instrument uncertainty. Therefore, the TS described levels in TS 2.2.7 and 2.2.8 did not adequately account for SIRWT instrument level uncertainty. As a result, using the levels described in TDB-III.40 for compliance with TS 2.2.7 and 2.2.8 was non-conservative.

The analysis concluded that there was inadequate/incomplete procedural guidance for developing Administrative Limits used to protect TS Limits. This includes guidance for understanding how to evaluate and apply uncertainties when developing TS Administrative Limits.

SIRWT level was increased to a level that accounted for instrument uncertainty. TDB-III.40 has been modified to change the Administrative Limits to account for uncertainty.

4OA4 IMC 0350 Inspection Activities (92702)

NRC inspectors began implementation of IMC 0350 inspection activities, which included follow-up on the restart checklist items contained in Confirmatory Action Letter (CAL) 4-12-002 issued June 11, 2012. The purpose of these inspections is to assess the licensee's performance and progress in addressing its implementation and effectiveness of FCS's Integrated Performance Improvement Plan (IPIP), significant performance issues, weaknesses in programs and processes, and flood restoration activities. Because the majority of restart checklist items being addressed by the licensee were in progress during this inspection timeframe the inspections primarily focused on assessing the status of licensee actions.

Inspectors used the criteria described in baseline and supplemental inspection procedures, various programmatic NRC inspection procedures, and IMC 0350 to assess the licensee's performance and progress in implementing its performance improvement initiatives. Inspectors performed on-site and in-office activities, which are described in more detail in the following sections of this report. This report covers inspection activities from July 16 through August 18, 2012. Specific documents reviewed during this inspection are listed in the attachment.

The following inspection scope, assessments, observations, and findings are documented by CAL restart checklist item number.

.1 <u>Causes of Significant Performance Deficiencies and Assessment of Organizational Effectiveness</u>

Section 1 of the restart checklist contains those items necessary to develop a comprehensive understanding of the root causes of safety-significant performance deficiencies identified at Fort Calhoun Station. In addition, Section 1 includes the independent safety culture assessment with the associated root causes and findings. The integration of the assessments under Item 1.f identifies the fundamental aspects of organizational performance in the areas of organizational structure and engagement, values, standards, culture, and human behaviors that have resulted in the protracted performance decline and are critical for sustained performance improvement. Section 1 reviews also include an assessment against appropriate NRC Inspection Procedure 95003 key attributes.

.a Flooding Issue – Yellow Finding

Item 1.a is included in the restart checklist for the failure of Fort Calhoun Station to maintain procedures and equipment that protects the plant from the effects of a design basis flood. These deficiencies resulted in a yellow (substantial safety significance) finding.

(1) <u>Inspection Scope</u>

Item 1.a is included in the restart checklist because the licensee failed to maintain procedures and equipment that protects the plant from the effects of a design basis flood. These deficiencies resulted in a finding having yellow (i.e., substantial) safety significance. During the inspection period covered by this report, the NRC inspectors assessed, and will continue to assess during upcoming inspection periods, the licensee's root cause, extent of cause, and extent of condition evaluations related to the Yellow finding. In addition, the inspectors started to verify, and will continue to verify during upcoming inspection periods, that corrective actions are adequate to address the root and contributing causes.

The onsite activities included a site familiarization tour that included a containment entry; a walk-down of the intake structure; a table-top exercise of Abnormal Operating Procedure (AOP)-1. "Acts of Nature" Section I, "Flood"; an initial walk-down of pre-staged flooding equipment; interviews with personnel involved in the flooding recovery efforts; and observation of recovery effort meetings. The in-office activities consisted of reviews of documents associated with the recovery efforts, procedures associated with flooding mitigation strategies, system lesson plans, and condition reports.

(2) <u>Assessment</u>

The inspectors' review focused mainly on the adequacy of procedures that are associated with mitigation strategies for a design basis flood. The inspectors' tabletop exercise of AOP-1, Section I, "Flood" revealed that, as compared to the circumstances that surrounded the Yellow Finding in 2009, the licensee has completed noteworthy improvements to this procedure to mitigate flood. The inspectors noted that the licensee had many of the flood-mitigating equipment staged during the first overview walk-down.

Overall, based on limited inspections, the licensee flood protection program is showing improvement. However, during the initial walk-downs of flooding procedures, the inspectors had several observations that were pointed out to the licensee. For example, the licensee currently stores a sand pile on the west side of the plant. This sand pile would be used for sandbagging various flood-susceptible areas around the plant in case of a flood. The inspectors noted that there was some vegetation growth on it and when it was brought up to the site personnel, it was discovered that there is no maintenance program to continually preserve the pile. The licensee entered this issue into their CAP as CR 2012-11088.
The inspectors plan to perform more detailed walk-downs of the various flood-mitigating procedures in future assessment periods to assess the suitability of the procedures to combat a flood.

(3) <u>Findings</u>

No findings of significance were identified.

.b Reactor Protection System contact failure – White Finding

Item 1.b is included in the restart checklist for the failure of Fort Calhoun Station to correct a degraded contactor which subsequently failed, in the reactor protection system. These deficiencies resulted in a white (low to moderate safety significance) finding.

(2) <u>Inspection Scope</u>

The NRC inspected and will continue to inspect the root cause, extent of cause, and extent of condition related to the contactor failure and the associated process failures.

The on-site activities included a walk-down of the control room areas that house the four contactors; interviews and discussions with staff performing evaluations related to this significant issue, a review of programs and processes being improved that led to this event; and observation of conduct of recovery effort meetings. The in-office activities consisted of reviews of documents associated with the recovery efforts, conditions reports, root cause analyses, scoping procedures, calculations, and drawings.

(2) <u>Assessment</u>

The team reviewed Revision 2 of the Root Cause Analysis for the contactor failure, RCA 2011-0451. The inspectors were informed that efforts were underway to revise this root cause analysis and issue Revision 3. Revision 3 was not completed during this assessment period and according to the licensee this revision is not expected to be completed until December of 2012. According to the management team at the station, revision 2 of RCA 2011-0451 had lots of errors and was being completely redone. The NRC will continue to follow the licensee actions in regards to this issue.

(3) <u>Findings</u>

No Findings of significance were identified.

.c Electrical Bus Modification and Maintenance – Red Finding

Item 1.c is included in the restart checklist for the failure to adequately design, modify, and maintain the electrical power distribution system, resulting in a fire in the safety-related 480 volt electrical switchgear. These deficiencies resulted in a red (high safety significance) finding.

(1) <u>Inspection Scope</u>

Item 1.c is included in the restart checklist because the licensee failed to adequately design, modify, and maintain the electrical power distribution system, which caused a fire in the safety-related 480 volt (V) electrical switchgear. These deficiencies resulted in a finding having red (i.e., high) safety significance. During the inspection period covered by this report, the NRC assessed, and will continue to assess during upcoming inspection periods, the licensee's root cause, extent of cause, and extent of condition evaluations related to the fire and associated equipment and process failures.

The on-site activities included a walk-down of the remains of the breaker fire event and a tour of the switchgear rooms; observing a demonstration of racking in a breaker; interviews and discussions with staff performing evaluations of significant performance issues, programs, and processes; and observation of recovery effort meetings. The in-office activities consisted of reviews of documents associated with the recovery efforts, conditions reports, root cause analyses, scoping procedures, calculations, and drawings.

(2) Assessment

Following the June 7, 2011, 1B4A breaker fire event, the licensee conducted two root cause analyses related to the event: CR 2011-5414, "Breaker Cubicle 1B4A Fire," which the licensee initiated on June 9, 2011, and CR 2011-6621, "1B3A Main Breaker Trip during Switchgear Fault on 1B4A," which the licensee initiated on September 12, 2011. CR 2011-5414 documents FCS's review of the event and focuses on the causes of the breaker fire. CR 2011-6621 documents the licensee's review of the breaker coordination issues and unexpected electrical distribution system response (i.e., the unexpected Breaker 1B3A trip) during the fire event on June 7, 2011. However, the NRC inspector observed that there was no evaluation or other corrective action program product that encompassed the entire event and that several significant conditions adverse to quality (SCAQs) had not been adequately addressed.

The inspectors conducted an independent review of the event using the guidance in NRC Inspection Procedure 95002, "Supplemental Inspection for One Degraded Cornerstone or Any Three White Inputs in a Strategic Performance Area," and focused on assessing the adequacy of the licensee's root and contributing causes for the event. The inspectors noted the following SCAQs during their review of the event:

1. A high impedance connection between the breaker cradle assembly and the 480 V bus stabs caused localized overheating and the bus bar failure, which initiated the event. This condition was the focus of CR 2011-5414. Corrective actions developed included replacing the damaged switchgear components, correcting and/or verifying the alignment of the remaining breaker and cradle assemblies, silver plating all the breaker stabs, and revising design procedures. The inspectors reviewed the corrective actions completed and planned and concluded they were adequate to preclude recurrence of this SCAQ.

2. During the fire, a phase-to-phase arc fault occurred for 42 seconds, which generated a fault current value of 16,000 amperes (A), until operators manually de-energized transformer T1B-4A by opening Breaker 1A4-10. In accordance with system design criteria and IEEE Standards, a fault should be isolated by the breaker closest to the fault. This would have isolated and arrested the fault and prevented it from impacting other busses. However, Breaker 1A4-10's breaker trip setpoint was such that a phase-to-phase fault on the line side of Breaker 1B4A would not be cleared. This allowed the fire to continue and produce combustion products and develop the subsequent fault across the BT-1B4A breaker. Although the licensee generated CR 2012-01630 on March 1, 2012, which acknowledged this condition, the licensee had yet to analyze the

adequacy of the breaker trip set points as of the conclusion of this inspection period.

3. The bus separation scheme was inadequate to meet the system's design criteria, IEEE standards, and the 1971 NRC Standard Review Plan (SRP). (Note: OPPD was licensed prior to the SRP). OPPD's scheme allowed combustion products from the 1B4A fire to be communicated to and affect bus tie breaker BT-1B4A because of the physical configuration of the bus duct work and because there is only one bus tie breaker on each end of the island busses. This configuration and the fire event resulted in the development of an electrical short between Bus 1B4A and Island Bus 1B3A-4A, which was powered from the opposite safety bus (Bus 1B3A). Thus both independent trains of vital AC power were adversely affected by a fault on a single bus. The licensee had not addressed this design deficiency as of the conclusion of this inspection period.

4. The bus separation scheme for the DC buses was inadequate. During the fire event on June 7, 2011, grounds developed on both DC buses. While a design basis fire is expected to impact one DC bus, both DC buses should not be impacted by a single fire. A loss of both DC buses would cause the loss of control power to all vital breakers. The licensee had not addressed this design deficiency.

5. The breaker coordination scheme did not respond as expected during the fire event. Breaker 1B3A tripped when a fault developed on Island Bus 1B3A-4A, which resulted in both Bus 1B3A and Island Bus 1B3A-4A being lost during the event. In accordance with system design requirements, Breaker BT-1B3A should have isolated the fault. Because of the fire and breaker coordination failure, six of nine vital 480 V buses were either manually or automatically de-energized during the event, and minimum ECCS system capacity was not maintained. This condition was the focus of CR 2011-6621. This SCAQ is discussed in greater detail in Section 4OA4.1.c(3) of this report. Corrective actions developed were reviewed by the inspectors and determined to be adequate to preclude repetition of this SCAQ.

6. The licensee did not initially adequately evaluate the safety significance of the fire event on June 7, 2011. The licensee concluded in its initial risk assessment, [CR 2011-5414-01 RE], dated June 24, 2011, that the event did not represent a nuclear safety risk. The licensee revised this assessment on September 12, 2011, and appropriately concluded the event represented a nuclear safety risk because if the event had occurred at power, minimum ECCS capacity would have been lost. Specifically, the event would have resulted in the loss of availability of high pressure make-up water (i.e., high pressure safety injection and charging) sources. This delay in accurately assessing the risk delayed the evaluation, troubleshooting, and corrective actions, and contributed to why all the SCAQs have not been identified and thoroughly evaluated.

Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), Appendix B, Criterion XVI, “Corrective Actions,” requires, in part, that measures established for identifying and correcting SCAQs shall also assure that the cause of the condition is determined and corrective action is taken to preclude repetition. This criterion also requires that the identification of the SCAQ, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management. The NRC does not consider this restart checklist item to be satisfied in part because the licensee has not demonstrated compliance with 10 CFR 50, Appendix B requirements for this significant performance deficiency. OPPD issued CR 2012-10625, to capture the inspectors’ concerns.

The team reviewed the license’s identified root and contributing causes and the corrective actions developed to address these causes. NRC assessments and observations, which will continue in subsequent inspection periods, are documented as follows.

<u>Root Cause: CR 2011-5414, “Breaker Cubicle 1B4A Fire,” Revision 2</u>

The licensee performed this root cause analysis (RCA) to determine what created the fire in the West Switchgear Room and the subsequent loss of 480 V Bus 1B4A. The licensee is in the process of developing the third revision of this RCA. The inspectors concluded thus far that the root and contributing causes identified in Revision 2 of CR 2011-5414 and the corrective actions proposed to address these causes were appropriate. The primary root cause was that the design process failed to identify critical parameters and interfaces. The licensee also identified nine contributing causes, which included engineering staff overreliance on vendor support, weaknesses in design procedures and checklists, lack of accessibility to the bus side of the switchgear, failure to take action when the acrid odor was identified, and inadequate post-maintenance testing. However, the inspectors identified additional contributing causes. These included an inadequate 10 CFR 50.59 review, pre-installation field walkdowns that failed to identify the physical size difference between the GE breakers and the NLI breaker finger assemblies, and quality control processes and procedures that failed to (1) require a detailed receipt inspection of the new safety-related equipment and (2) identify that the vendor did not provide drawings containing the sizes of the critical part.

URI 05000285/2011014-02 (ADAMS Accession No. ML12072A128) contains additional details about the 10 CFR 50.59 issue.

The inspectors concluded that the corrective actions were generally appropriate. However, the inspectors identified an instance in which a corrective action did not address a failure of the licensee’s CAP. One of the contributing causes of the event was that personnel did not adequately communicate the presence of an acrid odor that existed for three days preceding the event to engineering, maintenance, or management. Licensee personnel that identified the odor did not generate a condition report until two days after noticing the odor, and the CR that was generated was characterized as a “Class D” CR (i.e., not a condition adverse to quality). No actions were taken to evaluate the issue. Although the licensee is developing guidance for investigating an acrid odor, the NRC inspectors discussed concerns that the guide will not be effective if the organization is not immediately alerted to the existence of such conditions in the future. Additionally, the inspectors noted that the licensee did not generate a corrective action associated with the delay in entering the issue into the CAP over the 3 days the abnormal condition existed.

This RCA also documented the licensee's extent of condition and extent of cause reviews. The licensee appropriately identified the equipment subject to these reviews. However, the NRC identified that the timeliness for the extent of condition review was inadequate. The RCA established corrective actions to de-energize the bus, clean the stabs (e.g., remove any hardened grease), and correct the finger stab alignment such that contact occurs with the silver plated contact surface. The licensee assigned a due date for these actions nearly six months after the event and four months after it was verified the condition existed by visual and borescope inspections of the in-service switchgear. Considering that the licensee determined this condition was the direct cause of the failure, the corrective actions were neither timely nor commensurate with the safety significance of the event.

Root Cause: CR 2011-6621, "1B3A Main Breaker Trip during Switchgear Fault on 1B4A"

The licensee performed this RCA to determine why the coordination between the main feeder breaker 1B3A and the bus tie breaker BT-1B3A did not function as designed for load center 1B3A. The licensee initiated this RCA on September 12, 2011, during the NRC's Special Inspection of the fire event on June 7, 2011. The licensee identified two root causes for the event. The first root cause was that the vendor was unaware of the effect of the full function test kit (FFTK) on the zone selective interlock (ZSI) functionality. This knowledge gap caused a failure to establish a functionality test that would ensure proper breaker performance. The second root cause was that the design change package (DCP) preparation procedures did not provide adequate guidance for determining whether design features of new components could adversely affect required performance characteristics, especially if the new components were not properly configured.

The inspectors had a differing view about the licensee's first root cause. Although the root cause statement may be factual, the inspectors believed it was irrelevant in this case. The licensee purchased the breaker assembly, trip unit, and FFTK and assumed responsibility for the installation and testing of the breakers upon delivery and receipt of the equipment at Fort Calhoun. The licensee did not request the vendor to provide a test to verify the ZSI function was disabled. Standard factory testing and validation of customer setpoints was performed at the vendor site. The vendor manual for the trip unit and the FFTK, clearly states that the thermal imagining and ground fault detections functions are bypassed when using the FFTK and that special procedures are required to test the ZSI function and settings. While the vendor's knowledge of the FFTK can be considered a reasonable contributing cause, it cannot be considered a root cause because OPPD failed to request the vendor to provide a test to verify the ZSI function was disabled. OPPD never evaluated the new design functions of the NLI breaker assembly and Masterpack trip unit and was not aware of the adverse impact of the ZSI function. The inspectors believe a more appropriate root cause was that OPPD's quality assurance department and field technicians failed to verify that the safety-related equipment received from the vendor was properly configured. OPPD quality assurance department failed to require a separate receipt inspection to verify the equipment supplied by the vendor was configured correctly, met the purchase order specifications, and had installation instructions for verifying proper wiring configuration.

CR 2011-6621 identified four contributing causes:

1. Detailed standards for performing and documenting wire/continuity checks for new wiring do not exist. It is left to the test and field engineer to judge the level of detail required.
2. The design engineer did not properly employ the human performance toolbox in regard to maintaining a questioning attitude about the details of operation of the new breakers.
3. The field engineer and electricians did not properly employ the human performance toolbox in that they did not question the lack of detail in the CWO for performing wire and continuity checks.
4. The vendor manual for the Masterpact breakers does not clearly state how ZSI, if not properly restrained, will impact breaker coordination.

The inspectors did not agree with the fourth contributing cause. During the inspectors' review of the vendor manual for the Masterpact breaker trip unit, the team identified that the manual contained a page describing the ZSI function and specifically discussed the impact of having ZSI enabled with no established communications. There were also four discrete notes that cautioned the use of ZSI and the impact on plant operations if ZSI was not properly configured. Other contributing causes identified by the team included an overreliance on vendors, OPPD's engineering staff's lack of knowledge about the equipment, and a 10 CFR 50.59 review that failed to consider several new failure modes introduced by design features of the new breaker assembly (see URI 50-285 2011014-02 for additional details). This issue was discussed in the root cause analysis; however, OPPD did not considerit to be a contributing cause.

The inspectors' review of the corrective actions to address the identified causes is still in progress.

The licensee documented its extent of condition and extent of cause reviews in CR 2011-6621, and the analysis of potentially affected equipment was reasonable. These reviews were completed in a reasonable time. However, the team had concerns with the quality of the root cause analysis and the timeliness of corrective action activities.

For example:

- CRs 2011-5414 and 2011-6621 concluded significantly different values for the fault current during the event. CR 2011-6621 stated the fault current on the

1B4A was 4000 to 7000 amps, while CR 2011-5414 stated the fault current was 16,000 amps. OPPD confirmed that 16,000 amps was the correct fault current.

- The timeline for the event in CR 2011-6621 stated that the operators in the main control room attempted to open Breaker 1B4A, and this deenergized the 1B4A bus; however, Breaker 1B4A was found in the closed position. The 1B4A bus was not deenergized until the 1A4-10 breaker was opened.

- Breakers 1B3A and BT-1B3A were not removed from service and tested at the vendor's facility until March 2012. During this testing, the wiring error was discovered. Breaker 1B3A was in service for an additional 9 months. As a result, the plant's response to another fire (i.e., a design basis event) could have been identical.

The team's review of CR 2011-6621 was used to review and close out unresolved item (URI) 05000285/2011014-03, which was documented in Inspection Report 05000285/2011014. The URI is closed to the violation discussed in Section 4OA4.1.c(3) of this report.

This restart checklist item remains open.

(3) Findings

Introduction: The team identified a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." Specifically the design modification package for the 480 volt, alternating current (VAC) breaker replacements failed to ensure the breaker coordination for the 480 VAC electrical buses was maintained. As a result, feeder breaker 1B3A tripped unexpectedly during the fire in the 1B4A switchgear. This performance deficiency also resulted in the loss of multiple buses on both trains of 480 VAC, including ECCS systems, from a single fault on a 480 VAC bus.

Description: This issue was previously discussed in Inspection Report 05000285/2011014 as unresolved item (URI) 05000285/2011014-03, during the NRC Special Inspection of the 1B4A fire. During the fire event in the 1B4A switchgear on June 7, 2011, the feeder breaker to the 1B3A switchgear tripped unexpectedly, which de-energized a redundant train of safe shutdown equipment. The licensee performed a root cause analysis of the events associated with the fire in switchgear 1B4A and originally concluded that breaker 1B3A tripped on overcurrent based on inspection of the breaker following the event; however, additional investigations could not confirm this conclusion. OPPD initiated a separate Condition Report (CR 2011-6621) on September 12, 2011 and root cause analysis to investigate the breaker coordination aspect of the 1B3A breaker trip.

Six safety-related feeder breakers and six safety-related bus-tie breakers had been replaced in November 2009 in accordance with permanent plant modification EC 33464. The modification replaced General Electric AK-50 low voltage power circuit breakers with Nuclear Logistics Incorporated/Square-D Masterpact circuit breakers, cradle assemblies, and digital trip devices. The 480 VAC electrical distribution system is comprised of nine load centers; three load centers are fed from the 4160 VAC bus 1A3, and three load centers are fed from 4160 VAC bus 1A4. There are three island buses that can be energized from either 480 VAC bus via bus-tie breakers.

OPPD committed to meeting the criteria in IEEE 384-1981, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits.” This standard describes independence requirements for Class 1E equipment, including those required for safe shutdown. Section 5.10.1 of IEEE 384-1981 states that an electrically generated fire in one Class 1E division shall not cause a loss of function in its redundant Class 1E division. OPPD also committed to the design criteria in IEEE 308-1974, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.” Criterion 5.2.2(3), “Independence,” states that distribution circuits to redundant equipment shall be physically and electrically independent of each other. Criterion 4.6, “Equipment Protection,” states that Class 1E power equipment shall be physically separated from its redundant counterpart or mechanically protected as required to prevent the occurrence of common failure modes due to design basis events. The IEEE standard defines design basis events to include postulated phenomena such as fires.

To meet the above design criteria, the 480 VAC electrical distribution system is designed so that an electrical fault on one of the island busses should not affect the main 480 VAC bus because the normally-closed bus-tie breaker should selectively trip and isolate the fault from the other busses as described in Calculation EC-91-084, “Breaker and Fuse Coordination Study,” Revision 8. The bus-tie breakers have electronic trip settings with time-overcurrent trip values coordinated with those of the bus feeder breakers. These design requirements were not met because of a lack of adequate breaker coordination.

In EC 33464, OPPD staff erroneously assumed that the Nuclear Logistics Incorporated/Square-D Masterpact circuit breakers, cradle assemblies, and digital trip devices were a “like for like” equivalent replacement for the General Electric AK-50 low voltage power circuit breakers. As a result of this incorrect assumption, several new design features and physical differences between the breaker assemblies were not appropriately evaluated and introduced new failure modes. These design features included the use of WAGO connector blocks in the breaker internal wiring and digital trip unit features, such as thermal imaging and the zone selective interlock (ZSI). The ZSI feature of the trip unit allows the breaker to communicate with other breakers and use that communication to achieve breaker coordination. However, if no communications are established and ZSI is enabled, the breaker will trip instantaneously on a short-term fault, regardless of the setting of the short-term time delay setting. The ZSI feature is disabled by the installation of jumpers in the WAGO block assemblies, which is the normal factory (i.e., NLI) default setting.

In March 2012, feeder and bus-tie breakers 1B3A and BT-1B3A were removed from service and shipped to Nuclear Logistics, Inc. (NLI), for testing at the vendor’s facility. During these tests, the breakers were connected in series and a 9000 amp fault current was applied. Similar to the fire event on June 7, 2011, breaker 1B3A tripped instantaneously (0.06 sec), and the BT-1B3A breaker did not trip. Inspection of the breaker 1B3A internal wiring revealed that the jumpers in the WAGO block assembly were not in the required position; rather, they were offset by one row. As a result, the ZSI function was not disabled. When the proper jumper configuration was restored and the test re-performed, breaker BT-1B3A tripped to clear the fault current (0.28sec), and breaker 1B3A did not trip, which is the breaker coordination scheme required in the FCS design basis. Additional tests confirmed that the abnormal breaker response was caused by this configuration error.

Additional investigation revealed that primary injection testing of the new breaker assemblies was not sufficient to confirm that the ZSI function was disabled. When testing is done with the full function test kit (FFTK), the WAGO blocks, which contain the ZSI jumpers, are bypassed in order to disable the thermal imaging and ground fault functions. The vendor instruction manual states that special test procedures are needed to test the ZSI function; however, it does not explicitly indicate that the ZSI function is bypassed when using the FFTK.

The vendor's installation and manufacturing testing procedures required both vendor factory technicians and QA personnel to verify breaker wiring configuration prior to shipping. OPPD's installation procedure required OPPD field technicians to inspect the internal wiring and connections for tightness and to verify internal wiring configuration using the wiring diagram, which reflected the proper jumper configuration. All of these checks were completed and signed. When interviewed, the OPPD field technician could not remember specifically verifying the jumper configurations during wiring verification step and stated they did not check configuration when checking for connection tightness. In addition, the OPPD QA department did not conduct receipt inspections for these safety related breakers. No other records of work on the 1B3A breaker could be found that indicated that the jumper configuration was altered. OPPD's root cause investigation did not interview the vendor personnel who conducted these checks.

OPPD's root cause identified three failure mechanisms which could result in the ZSI feature being enabled: 1) mis-configuration of the ZSI jumpers on the WAGO block; 2) the WAGO block assembly becoming disconnected during the cradle racking process; and 3) damage to the cradle racking bar preventing pin engagement of the WAGO blocks. Examples of these failure mechanisms were discovered to have occurred during the November 2009 installation. However, only the jumper mis-configuration on breaker 1B3A impacted breaker ZSI function. OPPD also determined that the instructions for the technicians conducting the breaker wiring inspections contained insufficient details to ensure that the inspection verified the jumper configurations.

OPPD concluded that the most likely causes of the jumper mis-configuration were that the breakers were received from the vendor with the improper configuration and that the wiring inspections by the vendor and OPPD failed to identify the nonconforming condition. However, because of the lack of adequate documentation of receipt inspection and testing, OPPD could not definitively conclude these were the causes. OPPD entered this issue into its corrective action program as CR 2011-6621 and conducted a root cause evaluation and extent of condition review of the other 11 breakers to verify proper jumper configuration. Additional corrective actions including revisions to engineering and maintenance procedures have also been completed.

Analysis: The failure to ensure that the 480 VAC electrical power distribution system design requirements were properly implemented and maintained through proper modification, maintenance, and design activities contributed to causing a catastrophic fire in a switchgear that adversely impacted the required safe shutdown capability of the plant when the breaker coordination scheme did not perform as designed. This was a performance deficiency that was within OPPD's ability to foresee and prevent. The performance deficiency was evaluated using NRC Inspection Manual Chapter 0612, Appendix B, "Issue Screening," and the issue was determined to be more than minor because it affected the Initiating Events Cornerstone attributes of both the protection against external events attribute (i.e., fire) and the design control. The issue adversely affected the associated cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

The significance of this finding is bounded by the significance of a related Red finding regarding a fire in the 480 VAC safety-related switchgear in June 2011 (Inspection Report 05000285/2012010). The performance deficiency had a cross-cutting aspect in the area of human performance associated with resources (i.e., procedure accuracy) because OPPD failed to ensure that station procedures for engineering changes, plant modifications, inspections, installations, and maintenance contained sufficient details. [H.2(c)]

Enforcement: Title 10 CFR, Part 50, Appendix B, Criterion III, "Design Control," requires, in part that: (1) design changes, including field changes, be subject to design control measures commensurate with those applied to the original design; (2) measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions; and (3) these measures assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled.

Contrary to the above requirement, from November 2009 to March 2012, OPPD failed to: (1) assure that design changes were subject to design control measures commensurate with those applied to the original design; (2) assure that applicable regulatory requirements and the design basis for those safety-related structures, systems, and components were correctly translated into drawings, procedures, and instructions; and (3) assure that appropriate quality standards were specified and included in the design documents. Specifically, design reviews, work planning activities, and instructions for a modification to install new 480 VAC load center breakers failed to assure the breaker coordination scheme used to meet system design basis requirements was maintained when a new design feature was not adequately evaluated.

The licensee entered this issue into its corrective action program as CR 2011-6621. This violation is associated with a previous Red finding issued on April 10, 2012, regarding a significant internal fire event in the 480 VAC safety-related switchgear

(Enforcement Action 12-121). A separate citation will not be issued because this finding and its corrective actions will be managed by the Inspection Manual Chapter 0350 Oversight Panel. VIO 05000285/2012004-04, "Failure to Ensure Breaker Coordination of 480 VAC Electrical Power Distribution System Was Maintained."

.f Integrated Organizational Effectiveness Assessment

Item 1.f is included in the restart checklist because organizational effectiveness was identified as a potential key contributor to the overall decline in station performance. The NRC reviewed the licensee's root cause analysis of organizational effectiveness.

(1) Inspection Scope

Item 1.f is included in the restart checklist because organizational effectiveness was identified as a potential key contributor to the overall decline in station performance. The NRC reviewed the licensee's root cause analysis of organizational effectiveness and discussed the RCA and corrective actions with station personnel.

(2) Assessment

On June 9, 2012, FCS completed its RCA associated with Condition Report 2012-03986, "Organizational Ineffectiveness at Fort Calhoun Station Root Cause." The licensee performed this evaluation in part because of organizational effectiveness issues indicated by NRC Problem Identification and Resolution team inspections and an adverse trend in performance as indicated by the plant's movement in the NRC's ROP Action Matrix. The licensee used Conger & Elsea's MORT process to analyze its organizational effectiveness. The licensee concluded that the root causes for the decline in organizational effectiveness, which extended to programs, processes, and departments throughout the organization, were: (a) OPPD failed to establish and implement the essential attributes of governance and oversight, including the key elements of individual roles, responsibilities, and accountabilities to enable FCS to achieve and maintain high levels of operational nuclear safety and reliability; (b) station leaders were more tactical than strategic, prioritized poorly, delegated little, surrendered oversight, rationalized low standards, and hesitated to hold personnel accountable, resulting in a culture that valued harmony and loyalties over standards, accountability, and performance; and (3) OPPD leaders failed to develop, implement, and hold people accountable for implementation of important policies and programs, including the corrective action, operating experience, and observation programs.

The licensee's planned corrective actions included, in part: establishing governance, oversight, and workforce planning policies, processes, and programs; developing a strategic plan; and implementing management and accountability models. As of the conclusion of this inspection period, the licensee was developing an organizational effectiveness metric or performance indicator that will monitor the effectiveness of implemented corrective actions.

The team discussed its observations about the organizational effectiveness corrective action plans with FCS staff throughout the inspection period. The inspectors noted that the licensee concluded in its RCA that the station's programs and processes were adequate and that the implementation of those programs and processes were less than adequate. The team was concerned that this conclusion or assumption that programs and processes were adequate could be premature or incorrect because the licensee has not completed its reviews of and collective evaluations for the items in Section 3 of the restart checklist enclosed in the June 2012 CAL. In addition, the licensee was in the progress of identifying and developing root cause analyses for fundamental performance deficiencies (FPDs). One of these FPDs was associated with regulatory programs, which also contributed to the NRC's concern that the RCA assumption or conclusion was premature. The team also expressed its concern that the organizational effectiveness RCA results could potentially bias the conclusions and assumptions in the licensee's other efforts and RCAs. The licensee stated that it would perform an integrated review of the various ongoing assessments. NRC staff also discussed its concerns about the effectiveness and implementation of the organizational effectiveness RCA corrective actions and strategic planning documents not mentioning safety.

This restart checklist item remains open.

(3) Findings

No findings or violations of NRC requirements were identified; however, the NRC will continue its assessment of this CAL item as the licensee progresses in its implementation and monitoring of its corrective actions.

.2 Flood Restoration and Adequacy of Structures, Systems, and Components

Section 2 of the Restart Checklist contains those items necessary to ensure that important structures, systems, and components affected by the flood and safety significant structures, systems and components at Fort Calhoun Station are in appropriate condition to support safe restart and continued safe plant operation. Section 2 reviews will also include an assessment of the NRC Inspection Procedure 95003 key attributes as described in Section 6.

.a Flood Recovery Plan Actions Associated With Facility and System Restoration

Item 2.a is the NRC's independent evaluation of Fort Calhoun Station's Flood Recovery Plan. An overall flood recovery plan is important to ensure the station takes a comprehensive approach to restoring the facility structures, systems, and components to pre-flood conditions.

On August 30, 2011, Fort Calhoun Station issued Revision 1 to the "Fort Calhoun Station Post-Flooding Recovery Action Plan," (FRAP) that provided for extensive reviews of plant systems, structures, and components to assess the impact of the floodwaters. On September 2, 2011, the NRC issued Confirmatory Action Letter (CAL) 4-11-003, listing

235 items described in the Fort Calhoun Station Post-Flooding Recovery Action Plan that the licensee committed to complete. These 235 items were broken down into three sections: items to complete prior to exceeding 210 degrees Fahrenheit in the reactor coolant system, items to complete prior to reactor criticality; and items to complete following restart of the plant. On June 11, 2012, the NRC issued CAL 4-12-002. This CAL incorporates all the actions required by CAL 4-11-003.

The areas to be inspected are identified in the CAL. Inspection items are considered complete when the licensee has submitted a closure package that has been satisfactorily reviewed by the inspectors.

(1) CAL Action Items 2.3.1.13, 2.3.1.14, 2.3.1.15, and 2.3.1.16

i. Inspection Scope

The purpose of Action Items 2.3.1.13, 2.3.1.14, 2.3.1.15, and 2.3.1.16, was to procure, remove, install, and perform postmaintenance testing on pump motors DW-69-M and DW-70-M (Reverse Osmosis Unit Water Storage Tank Inlet and Outlet Pump Motors). These items were required to be completed prior to exceeding 210 degrees Fahrenheit in the reactor coolant system.

The licensee procured two pump and motor assemblies to replace the degraded pumps. The licensee installed the new pump and motor assemblies on November 16, 2011, and completed postmaintenance testing on December 22, 2011.

The inspectors reviewed the procurement documents to ensure that the replacement pump and motor assemblies were satisfactory replacements. The inspectors also witnessed postmaintenance testing and reviewed the postmaintenance testing data to ensure the pump and motor assembly installations were adequate.

This activity constitutes completion of Action Items 2.3.1.13, 2.3.1.14, 2.3.1.15, and 2.3.1.16 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(2) CAL Action Items 5.1.2.3, 5.1.2.5, and 5.1.2.7

i. Inspection Scope

The purpose of these action items were to identify and correct Alert and Notification System infrastructure issues, and return the system to its approved design configuration. These items were required to be completed prior to reactor criticality.

The inspector performed in-office and on-site reviews of the closure documentation for Action Items 5.1.2.3, 5.1.2.5, and 5.1.2.7. The inspector:

- Reviewed the Recovery Action Closure Verification Checklist for returning the Alert and Notification siren system to functional status (Flood Recovery Plan Item 5.1.2.3), dated May 7, 2012;

- Reviewed licensee procedures EPT-1, "Alert Notification System Silent Test," Revision 17, and EPT-2, "Alert Notification System Growl Test," Revision 18;

- Reviewed Siren Maintenance Checklists for sirens 1, 69, 75, 76, 135, 143, 257, 259, and 260;

- Reviewed Alert and Notification System performance indicator data submitted by the licensee for the period June 2011 through June 2012; and,

- Toured offsite emergency warning sirens 75, 76, and 135, on March 28, 2012

This activity constitutes completion of Action Items 5.1.2.3, 5.1.2.5, and 5.1.2.7, as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(3) CAL Action Item 5.1.3.1

i. Inspection Scope

The purpose of verifying the condition of informational signs for transient members of the public was to ensure that persons using nearby recreational areas are informed about emergency signals and sources of emergency information in the event of an emergency at Fort Calhoun Station. This item was required to be completed prior to reactor criticality.

The inspector conducted an in-office review of the closure documentation for Action Item 5.1.3.1, including,

- The Recovery Action Closure Verification Checklist for verifying emergency preparedness informational signs for transient members of the public, dated June 21, 2012;

- Procedure EPT-37, "Verification of Warning Signs," Revision 20;

- Records for sign verifications performed on September 20 and November 17, 2011, and March 14, 2012; and

- Work Orders 00412108, dated September 20, 2011, and 00429513, dated November 17, 2011 The inspector also observed four informational signs located in the DeSoto Bend National Wildlife Refuge on March 28, 2012.

The inspector noted that three of twenty-eight signs were verified as present and readable on September 15, 2011, ten signs were verified on November 9, 2011, and that all twenty-eight signs were verified on March 14, 2012. Some signs were located in areas not open to the public on the dates the verifications were performed.

This activity constitutes completion of verifying emergency preparedness informational signs action item, as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(4) CAL Action Item 5.1.2.8

i. Inspection Scope

The purpose of conducting a full-system test of the offsite Alert and Notification System was to verify the system was fully operational and had been returned to its approved configuration. This item was required to be completed prior reactor criticality.

The inspector performed an in-office review of the closure documentation for Action Item 5.1.2.8, including:

- The Recovery Action Closure Verification Checklist for Conduct a Full Siren System Test, dated May 14, 2012;

- Procedure EPT-3,"Alert Notification System Complete Cycle Test," Revisions 15 and 16; and,

- Test results for the Complete Cycle Test conducted October 26, 2011

ii. Findings

The inspector determined that four sirens had not been returned to service prior to the Complete Cycle Test conducted October 26, 2011. Because a full-system test of the offsite Alert and Notification System has not been completed in its as-designed configuration, CAL Action Item 5.1.2.8 remains open.

(5) CAL Action Item 5.3.2.2

i. Inspection Scope

The licensee is required by the Fort Calhoun Station Radiological Emergency Response Plan, Section N.2.1.1, dated March 3, 2005, to perform monthly surveillances of the primary and backup communications systems used to communicate with State and local governments. The purpose of these surveillances is to ensure the licensee's capability to notify offsite authorities of emergency conditions and to provide appropriate event information to offsite responders. This item was required to be completed prior reactor criticality.

The inspector performed an in-office review of the closure documentation for Action Item 5.3.2.2, including:

- Recovery Action Closure Verification Checklist for Flood Recovery Plan Item 5.3.2.2, dated May 15, 2012;

- Procedures EPT-5, "EOF Federal/State Communications Check," Revision 30, and EPT-6, "TSC/CR Federal/State Communications Check," Revision 25; and,

- Results for communication equipment tests conducted between June and October 2011;

This activity constitutes completion of the Perform normal Communications System testing action item, as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(6) CAL Action Items 5.3.2.3, 5.3.2.4, and 5.3.2.5

i. Inspection Scope

The purpose of verifying the operability of plant instrumentation used in the Emergency Plan is to ensure the licensee's capability to promptly recognize abnormal plant conditions requiring declaration of an emergency condition. These items were required to be completed prior reactor criticality.

The inspector verified that the plant radiation monitoring system, plant effluent monitors, and other plant instruments required by the site Emergency Plan were operable during onsite inspections conducted October 19-20, 2011, and March 26-30, 2012.

This activity constitutes completion of the verify the operability of plant instrumentation used in the Emergency Plan action item as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(7) CAL Action Items 5.3.2.18 and 5.3.2.19

i. Inspection Scope

The Federal Emergency Management Agency monitored the condition of offsite emergency preparedness throughout the flooding event at Fort Calhoun Station, June through September 2011. Federal Emergency Management Agency Region VII conducted a Disaster Initiated Review checklist to assess the station and offsite agencies' ability to respond to an actual event. These items were required to be completed prior reactor criticality.

The inspector performed an in-office review of the closure documentation for Action Items 5.3.2.18 and 5.3.2.19, including:

- The Recovery Action Closure Verification Checklist for Obtaining a Statement of Reasonable Assurance, dated July 3, 2012; and,

- The Statement of Continuation of Reasonable Assurance for Fort Calhoun Nuclear Station, dated November 22, 2011

This activity constitutes completion of the Obtain a Statement of Reasonable Assurance action item, as described in Confirmatory Action Letter 4-12-003.

ii. Findings

No findings were identified.

(8) CAL Action Item 5.4.2.4

i. Inspection Scope

The purpose of conducting a thorough critique of emergency preparedness performance during the Missouri River flooding event is to identify inadequate emergency response organization programs, processes, equipment, procedures, and training. The correction of identified weaknesses will improve station performance during future events. This item was required to be completed prior reactor criticality.

The inspector performed an in-office review of the closure verification checklist and supporting documentation for Action Item 5.4.2.4, including,

- Recovery Action Closure Verification Checklist for the critique of emergency preparedness performance during the Missouri River flooding event, dated July 3, 2012; and,

- Post Event Report: “NOUE – Flooding, June 5 through August 29, 2011, NRC Event Number #46929,” dated December 1, 2011. This activity constitutes completion of the Critique the Missouri River flooding event action item, as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

.3 **Adequacy of Significant Programs and Processes**

Section 3 of the Restart Checklist addresses major programs and processes in place at Fort Calhoun Station. Section 3 reviews will also include an assessment of how the licensee appropriately addressed the NRC Inspection Procedure 95003 key attributes as described in Section 6.

.a Corrective Action Program

(1) Inspection Scope

The Corrective Action Program and the use of industry Operating Experience at a nuclear power plant is a key element in ensuring the licensee's ability to effectively detect, correct, and prevent problems. A properly functioning Corrective Action Program is also a basis for licensee operation within the Reactor Oversight Process. Based upon observed problems with Corrective Action Program effectiveness, in both multiple examples of significant findings and identified issues in an NRC problem identification and resolution inspection, the licensee is performing a comprehensive review of this program.

The NRC will assess the licensee's review and potential changes to the Corrective Action Program. The NRC will also conduct independent inspections to validate whether the Corrective Action Program is appropriately functioning.

For the assessment period covered by this inspection report, the onsite activities included a Corrective Action Program presentation that described the life of a condition report and a discussion on each of the different phases to disposition an issue. This included a description of the different processes and site departments and personnel in charge of identification, screening, evaluation and resolution of an issue. In addition, the presentation also familiarized the inspectors with the Action Way system, where the licensee tracks progress of open condition reports. The inspectors also observed CAP meetings such as Station Corrective Action Review Board (SCARB), Condition Review Group (CRG), and Daily Screening Team (DST). Lastly, the inspectors interviewed site personnel associated with the Performance Improvement department to get a better understanding of the site processes. The in-office activities, which were conducted at the inspectors' regular duty stations, consisted of reviews of root cause analyses and procedures associated with the Corrective Action Program.

(2) Assessment

During this assessment period, the inspectors focused mainly on understanding the different processes and phases of the Corrective Action Program. The inspectors attended two CRG and two SCARB meetings and one DST meeting. To be able to reasonably assess these processes, the inspectors will attend more of these meetings and observe more of the CAP processes during future on-site inspection weeks. In general, the inspectors noted a general attitude to follow the CAP procedures and healthy willingness to express dissenting views during CAP meetings. Overall the inspectors noted that the Corrective Action Program procedures appeared appropriate and that the site would have a successful program if they were consistently followed.

(3) Findings

No findings of significance were identified.

.b Equipment Design Qualifications

This item of the Restart Checklist determines whether plant components are maintained within their licensing and design bases. Additionally, this item provides monitoring of the capability of the selected components and operator actions to perform their functions. As plants age, modifications may alter or disable important design features making the design bases difficult to determine or obsolete. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully.

.i Safety-Related Parts Program

A number of instances have been identified where non-safety-related parts have been installed into safety-related applications. Fort Calhoun Station is performing reviews to identify conditions where a non-safety-related component or subcomponent was improperly used in a safety-related application. The restart checklist includes an NRC assessment of the licensee's equipment design qualifications review for inconsistent quality classifications and the licensee's review of the use of non-safety-related parts in safety-related applications.

(1) Inspection Scope

NRC inspectors reviewed the licensee's procedure, scope of work, and training material for assessing their safety-related parts program. Inspectors also interviewed station personnel and contractors that performed the reviews. Inspectors reviewed a sample of the condition reports generated from the review and draft revisions of the individual system and collective evaluations, many which has not been finalized as of the end of the inspection period covered by this report.

(2) Assessment

During the inspection period, OPPD completed the discovery phase of its evaluations of this issue. The discovery phase was designed to identify all work orders (WOs) where non safety related parts were issued for jobs involving safety-related SSCs. This process identified 2100 WOs to be evaluated to determine if non safety related parts were installed in safety-related system and, if so, whether these parts impacted the system's functionality and operability. At the end of the inspection period, the licensee had reviewed 172 of the 2100 WOs, and two of those WOs required an evaluation of the impact on system functionality and operability. The NRC inspectors will continue to review all instances of WO issues that resulted in system functionality evaluations, and the team will assess a sampling of the WOs for which further evaluations were performed to determine the effectiveness of the licensee's review. This restart checklist item will remain open until all WOs have been screened and questions related to operability of SSCs required for Modes 1 and 2 have been appropriately evaluated and addressed.

(3) Findings

No findings or violations of NRC requirements were identified; however, the NRC will continue its assessment of this CAL item.

.c Design Changes and Modifications

Modifications to risk-significant structures, systems, and components can adversely affect their availability, reliability, or functional capability. Modifications to one system may also affect the design bases and functioning of interfacing systems. Similar modifications to several systems could introduce potential for common cause failures that affect plant risk. A temporary modification may result in a departure from the design basis and system success criteria. Modifications performed during increased risk configurations could place the plant in an unsafe condition.

This item assesses the effectiveness of the licensee's implementation of changes to facility structures, systems, and components, risk significant normal and emergency operating procedures, test programs, evaluations required by 10 CFR 50.59, and the updated final safety analysis report. The NRC has future inspections planned to address this CAL item.

.i Vendor Modification Control

Past NRC inspections indicated that the licensee failed to ensure critical characteristics were identified and properly addressed in several modification packages. FCS is currently reviewing work performed by vendors. The restart checklist includes an NRC assessment of the effectiveness of the licensee's vendor program, including its oversight of vendor work.

(1) Inspection Scope

NRC inspectors reviewed the licensee's procedure, scope of work, and training material for assessing past vendor work packages. Inspectors also interviewed station personnel and contractors that performed the reviews. Inspectors reviewed a sample of the condition reports generated from the review and a draft revision of the collective evaluation, which has not been completed as of the end of the inspection period covered by this report.

(2) Assessment

The licensee evaluated vendor-prepared modifications from the past 5 years to determine whether critical characteristics were identified and properly addressed. Licensee personnel stated that this review focused on the design characteristics of the modifications. The licensee's procedure established criteria for expanding the timeframe and number of modifications reviewed if it identified problems of varying severity; however, the licensee stated that it did not identify any fundamental flaws that met the scope expansion criteria. The inspectors observed that the scope of the review did not cover other aspects of the licensee's vendor program, such as procurement, receipt inspection, and installation activities performed by the vendor.

The licensee generated 23 condition reports based on its review and revised its modification process procedures to incorporate guidance for capturing critical characteristics. The licensee was in the process of developing its collective evaluation of the issues it identified. The inspectors expressed a concern about the licensee's use of the term "administrative" to characterize the identified problems because the term was not defined in licensee procedures. Inspectors were concerned that the term could potentially cause personnel to incorrectly conclude that programmatic errors do not have an impact on nuclear safety.

The NRC will continue its review of the condition reports and a sample of the modification packages evaluated by the licensee. The NRC will also review the collective evaluation results, corrective actions, and the effectiveness of those corrective actions when completed by the licensee. This restart checklist item remains open.

(3) Findings

No findings or violations of NRC requirements were identified; however, the NRC will continue its assessment of this CAL item.

.ii 10 CFR 50.59 Screening and Safety Evaluations

Past NRC inspections indicated that several changes to the facility were not properly screened or evaluated in accordance with the requirements of 10 CFR 50.59. FCS is evaluating past 10 CFR 50.59 documents. The restart checklist includes an NRC assessment of plant and procedure modifications to determine if those modifications were appropriately evaluated in accordance with 10 CFR 50.59. The NRC will also evaluate the effectiveness of the licensee's 10 CFR 50.59 process to ensure proper treatment of changes to the facility.

(1) Inspection Scope

NRC inspectors reviewed the licensee's procedure, scope of work, and training material for assessing its 10 CFR 50.59 documents. Inspectors also interviewed station personnel and contractors that performed the reviews. Inspectors reviewed a sample of the condition reports generated from the review.

(2) Assessment

The licensee evaluated 100 percent (%) of its 10 CFR 50.59 evaluations, 20 % of its 10 CFR 50.59 screenings, and 20 % of its 10 CFR 50.59 applicability determinations to determine if any deficiencies areas for improvement existed in the technical adequacy of those documents. The licensee stated that it reviewed these documents, and approximately 28 condition reports were generated. The licensee's procedure established criteria for expanding the timeframe and percentage of documents reviewed if it identified problems of varying severity. The licensee stated that it did meet the scope expansion criteria. The licensee has not yet provided the NRC its collective evaluation of the issues it identified.

The team identified concerns with the scope of the licensee's review in this area. The licensee's initial scope of review did not include the 50.59 screening for the engineering calculation associated with the 480 VAC breaker replacement modification (EC 33464) despite this 50.59 screening being the subject of NRC URI 05000285/2011014-02 and included in the problem statement for this project. The licensee identified problems with other 50.59 documentation, which then triggered the scope expansion criteria. This 50.59 screening for EC 33464 was then included in the review on August 16, 2012 as part of the scope expansion. The licensee generated CR 2012-10391 to document deficiencies with the its 50.59 screening for this modification. In addition, the NRC inspectors expressed concerns that only 20% of the 50.59 screenings from the past five years was included in the scope of this review despite previous problems with 50.59 screenings.

The NRC will continue its review of the 50.59 documentation and associated condition reports evaluated by the licensee. The NRC will also review the collective evaluation results, corrective actions, and the effectiveness of those corrective actions when completed by the licensee. This restart checklist item remains open.

(3) Findings

No findings or violations of NRC requirements were identified; however, the NRC will continue its assessment of this CAL item.

.d Maintenance Programs

Inadequate maintenance activities that are not detected prior to returning the equipment to service can result in a significant increase in unidentified risk for the subject system.

The Maintenance Rule (10 CFR 50.65) requires licensees to monitor the performance or condition of structures, systems and components within the scope of the rule against licensee-established goals to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions. These goals are to be commensurate with safety and, where practical, should take into account industry-wide operating experience.

The NRC will assess the licensee's maintenance programs, including preventative maintenance, compliance with vendor recommendations, post-maintenance testing programs, and establishing and controlling equipment service life. The NRC will also inspect the licensee's qualifications and documentation to certify equipment for harsh environments (10 CFR 50.49).

(1) Inspection Scope

i. Vendor Manuals and Vendor Informational Control Programs

NRC inspections determined vendor manuals and information have not been adequately maintained, which resulted in adverse conditions at Fort Calhoun Station. The licensee is performing a review to identify and incorporate updates to vendor manual technical documentation. This review applies to all equipment and components classified as a Critical Quality Element (safety-related). Changes in vendor guidance will be evaluated to determine what impact, if any, the new information has on scheduled work, work completed since the last vendor manual update was made, and changes to plant documentation. The NRC will evaluate the effectiveness of the licensee's incorporation of vendor information into applicable plant procedures and design documents to ensure proper maintenance and operation of facility equipment.

ii. Equipment Service Life

NRC inspections determined that the licensee opted to keep some plant equipment in service beyond the vendor recommended service life or standard industry guidelines. Operating equipment past the recommended replacement timeline has resulted in age-related failures at Fort Calhoun Station. In response, the licensee will perform an assessment to evaluate the service life of safety-related plant equipment and the effectiveness of programs used to implement service life requirements. The NRC will inspect and assess the adequacy of this evaluation and the associated corrective actions.

(2) Assessment

The inspectors noted that the dates of completion and work in the licensee's schedule do not line up. Because the vendor manuals contain the service life requirements for most equipment and their subcomponents, it did not seem reasonable to the inspectors to complete the service life review prior to completing the vendor manual review without a reconciliation process to ensure that items were not missed. The inspector also noted that this issue may exist in other programs that depend on each because so many of these programs are inter-related. The licensee agreed and wrote Condition Report CR 2012-09215 to address this concern.

(3) Findings

No Findings of significance were identified.

.4 **Review of the Integrated Performance Improvement Plan**

Section 4 of the Restart Checklist is provided to assess Fort Calhoun Station's Integrated Performance Improvement Plan. OPPD will provide the Integrated Performance Improvement Plan, which details the plans and actions needed to address the conditions that transitioned FCS to NRC oversight under IMC 0350.

The Integrated Performance Improvement Plan (IPIP) should address pre-restart and post-restart actions. The IPIP should have a sufficient level of detail so that the NRC staff will be capable of developing inspections plans to assess and review the plan's actions. Additionally, OPPD should provide a nexus between the IPIP and NRC Inspection Procedure 95003.

The NRC will review the IPIP to ensure its pre-startup and post-startup actions and plans are adequate to address the conditions that led to the protracted decline in plant performance.

(1) Inspection Scope

The team reviewed revision 3 of the licensee's IPIP and interviewed licensee personnel involved with developing the IPIP. The team also reviewed FCS station performance indicators referenced in the IPIP.

(2) Assessment

The licensee stated that the IPIP combined several initiatives that were ongoing at FCS. The IPIP contains a vision statement, goals, a plan overview, and a listing of actions associated with flood recovery. The team expressed some concerns with the organization of the IPIP's contents and the clarity of the relationships among the vision statement, goals, and action plans. The licensee stated that revision 4 of the IPIP is in progress.

(3) Findings

No findings or violations of NRC requirements were identified; however, the NRC will continue its assessment of this CAL item.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On June 22, 2012, the inspectors presented the results of the radiation safety inspections to Mr. M. Prospero, Plant Manager, and other members of the licensee staff. On August 3, 2012, the lead inspector conducted a final exit of the inspection result, telephonically, with Mr. D. Bannister, Vice President and Chief Nuclear Officer. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On August 27, 2012, the inspectors presented the integrated inspection results to Mr. L. Cortopassi, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Acker, Licensing Engineer
D. Bannister, Vice President and Chief Nuclear Officer
S. Baughn, Manager, Nuclear Licensing
R. Beck, Supervisor, System Chemistry
B. Blome, Manager, Quality Assurance
D. Brehm, Supervisor, Rad-Equipment
C. Cameron, Supervisor Regulatory Compliance
K. Erdman, Supervisor, Programs
M. Ferm, Manager, Site Performance Improvement
M. Frans, Manager, Engineering Programs
J. Goodell, Division Manager, Nuclear Performance Improvement and Support
P. Gunderson, Supervisor, Radiation Protection
W. Hansher, Supervisor, Nuclear Licensing
R. Haug, Manager, Training
J. Herman, Division Manager, Nuclear Engineering
R. Hodgson, Manager, Work Management
E. Jun, System Engineer
A. Kelly, Chemist
K. Kingston, Manager, Chemistry
T. Maine, Manager, Radiation Protection
E. Matzke, Senior Licensing Engineer
S. Miller, Manager, Design Engineering
K. Naser, Manager, System Engineering
T. Nguyen, System Engineer
A. Pallas, Manager, Shift Operations
M. Prospero, Division Manager, Plant Operations
J. Shipman, Supervisor, Chemical Operations
M. Smith, Manager, Operations
T. Uehling, Manager, Maintenance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285/2012004-04	VIO	Failure to Ensure Breaker Coordination of 480 VAC Electrical Power Distribution System Was Maintained
05000285/2012-001-00	LER	Inadequate Flooding Protection Procedure

Opened

05000285/2012-002-00	LER	Inadequate Qualifications for Containment Penetrations Renders Containment Inoperable
05000285/2012-003-00	LER	Non-Conservative Error in Calculation for Alternate Hot Leg Injection Results in Unanalyzed Condition
05000285/2012-004-01	LER	Inadequate Analysis of Drift Affects Safety Related Equipment
05000285/2012-005-01	LER	TS violation due to inadequate testing of DG fuel pumps
05000285/2012-006-00	LER	Operation of Component Cooling Pumps Outside of the Manufacturers Recommendation
05000285/2012-007-00	LER	Failure of Pressurizer Heater Sheath
05000285/2012-008-00	LER	Technical Specification Violation for Fuel Movement (VA-66)
05000285/2012-009-00	LER	Inoperable Equipment due to Lack of Environmental Qualifications
05000285/2012-010-00	LER	Seismic Qualification of Instrument Racks
05000285/2012-011-00	LER	Emergency Diesel Inoperability Due to Bus Loads During a LOOP
05000285/2012-012-00	LER	Multiple Safety Injection Tanks Rendered Inoperable
05000285/2012-013-00	LER	Inadequate Calculation of Uncertainty Results a Technical Specification Violation

Opened and Closed

05000285/2012004-01	NCV	Failure to report an event to the NRC within 60 days for an operation prohibited by Technical Specifications
05000285/2012004-02	NCV	Fuel Move with SFP Ventilation Inoperable a Condition Prohibited by Technical Specification 2.8.3(4)
05000285/2012004-03	NCV	Failure to Establish and Implement Adequate Procedures for Meteorological Monitoring and the Off-Site Dose Calculation Manual
05000285/2012-004-00	LER	Inadequate Analysis of Drift Affects Safety Related Equipment
05000285/2012-005-00	LER	TS violation due to inadequate testing of DG fuel pumps

LIST OF DOCUMENTS REVIEWED

Section 1R07: Heat Sink Performance

Procedures

NUMBER	TITLE	REVISION / DATE
PE-RR-CCW-0100	Disassembly, Cleaning, and Repair of CCW Heat Exchanger – Raw Water Side	38
PBD-17	Service Water Reliability	5
PED-SEI-16	Evaluation of Heat Exchanger Performance	10
SE-PFT-CCW-0001	Component Cooling Water Heat Exchanger Performance Test	15
CH-AD-0035	Microbiologically Induced/Influenced Corrosion Monitoring Program	3
CH-AD-0048	Environmental Inspection for Biofouling Organisms	3

Drawings

NUMBER	TITLE	REVISION / DATE
11405-M-100	Piping and Instrumentation Diagram Raw Water Flow Diagram	100

Miscellaneous Documents

TITLE	REVISION / DATE
FCS Program Health Reports, Service Water Reliability	4Q2011-1Q2012
FCS System Health Reports, Auxiliary Cooling System	1Q2012
EPRI-NP-7552, Heat Exchanger Performance Monitoring Guidelines	12/1991

Condition Reports

2010-5680	2010-5772	2011-0077	2012-8770

Work Orders

396227	396368

Section 2RS06: Radioactive Gaseous and Liquid Effluent Treatment

PROCEDURES

NUMBER	TITLE	REVISION
CH-AD-0011	Sampling Guidelines	6
CH-AD-0029	Quarterly Cumulative Dose Calculations from Radioactive Effluents	6
CH-AD-0050	Annual Radioactive Effluent Release Report	9
CH-AD-0055	Special Radiological Liquid Release Permit and Summary	3
CH-AD-0060	Groundwater Sampling and Analysis Process	2
CH-ODCM-0001	Off-Site Dose Calculation Manual	21
CH-SMP-RE-0004	Atmosphere Sampling Radioactive Gas Particulate and Iodine Using Either RM-050/051 or RM-052	18
CH-SMP-RE-0013	Auxiliary Building Exhaust Stack Sampling	24
CH-SMP-RE-0018	Laboratory and Radioactive Waste Processing Building Exhaust Stack Sampling	25
CH-ST-MM-0001	Quarterly Determination of Doses from Liquid and Gaseous Releases	6
CH-ST-MM-0002	Annual Radiological Effluent Report	8
CH-ST-RM-4300	Laboratory and Radioactive Waste Processing Building Exhaust Stack Gas Radiation Monitor, RM-043, Primary Calibration	5

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
22937	NUPIC Audit of Teledyne Brown Engineering Environmental Services	February 17, 2011
2009-1219	FCS Self-Assessment Report: Chemistry Programs	May 2, 2010
10-QUA-083	SARC Audit Report No. 63 – Radiological Effluent Controls, Environmental Monitoring and Waste Management Programs, and ODCM	December 28, 2010
11-QUA-079	Quality Department Surveillance Report: Environmental Monitoring	September 30, 2011

CONDITION REPORTS

2010-02502	2010-05027	2010-05808	2011-01640	2011-03007
2011-05460	2011-07600	2011-07798	2011-07800	2011-09836
2011-10003	2011-10438	2012-01648	2012-03394	2012-03714
2012-03939	2012-04171	2012-04186	12-0451520	

RELEASE PERMITS

RELEASE NUMBER	SYSTEM	DATE
2011001	Component Cooling Water	April 23, 2011
2011002	Component Cooling Water	May 18, 2011
2011003	Component Cooling Water	May 25, 2011
2011004	Railroad Siding	July 8, 2011

2011005	Component Cooling Water	August 4, 2011
2011006	Component Cooling Water	September 22, 2011
2011090	“A” Monitor Tank	June 2, 2011
2011110	“B” Monitor Tank	July 28, 2011

IN-PLACE FILTER TESTING RECORDS

UNIT	SYSTEM	DATE
37606901	Control Room Charcoal Filter VA-64B	November 22, 2010
39673501	Spent Fuel Storage Pool Area Charcoal Filter VA-66	December 22, 2011
39944701	Safety Injection Pump Room Charcoal Filter VA-26A/B	February 14, 2012
39602701	Freon Test of Spent Fuel Pool Area Charcoal Filter VA-66	December 2, 2011
43385501	VA-64A Control Room HEPA and Charcoal Filter	January 14, 2012
41833501	Control Room Charcoal Filter VA-64A	January 14, 2012
40035801	Freon Test of SI Pump Room Charcoal Filter	February 14, 2012
41758501	VA-64B Control Room HEPA and Charcoal Filter	February 14, 2012

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION/DATE
	Updated Safety Analysis Report – Chapter 11.2: Radioactive Waste and Radiation Protection and Monitoring	17
	Updated Safety Analysis Report – Chapter 11.3: Radiological Effluent Requirements	6
107C-865972-B	Plant Diagram: Housing Assembly, VA-66, Carbon Cell (12,800 CFM) – Spent Fuel Pool	May 22, 1985
107C-865212-B	Plant Drawing: Filter Housing, Carbon, VA-26A&B – SI Pump Room	June 24, 1970
CHRF1103	Outage Primary Chemistry Refresher	2012
	System Health Report for Radiation Effluent Monitors	2009-2012
	Annual Radiological Effluent Release Report	2010
	Annual Radiological Effluent Release Report	2011

Section 2RS07: Radiological Environmental Monitoring Program

PROCEDURES

NUMBER	TITLE	REVISION
CH-AD-0011	Sampling Guidelines	6
CH-AD-0049	Annual Meteorlogical Data	5
CH-AD-0050	Annual Radioactive Effluent Release Report	9
CH-AD-0054	Annual Environmental Operating Report	EC 45174
CH-AD-0057	Siting Critiera and Sampling Strategies for Collection of Radiological Environmental Monitoring Samples	EC44353
CH-AD-0060	Groundwater Sampling and Analysis Process	2
CH-ST-RV-0001	Environmental Sample Collection - Water	EC 45736
CH-ST-RV-0002	Environmental Sample Collection – Milk or Equivalent	EC 45736
CH-ST-RV-0003	Environmental Sample Collection – Quarterly Environmental Dosimeters (TLDs)	14
CH-ST-RV-0004	Environmental Sample Collection – Sediment	EC 45736

CH-ST-RV-0005	Environmental Sample Collection – Fish	7
CH-ST-RV-0006	Environmental – Land Use Survey	EC 38421
CH-ST-RV-0007	Environmental Sample Collection – Vegetables or Food Products	EC 45736
CH-ST-RV-0008	Environmental Sample Collection – Air Monitoring	EC 45736
CH-ST-RV-0010	Environmental Monthly Progress Report Receipt	EC 41161
CH-ST-RV-0011	Environmental Sample Collection – Groundwater	EC 45004
CH-SMP-RV-0006	Quality Assurance Water Sample Composite (QA)	7
CH-SMP-RV-0012	Environmental Sample Shipment	2
CH-SMP-RV-0013	Environmental Vegetation Sample Collection	2
CH-SMP-RV-0014	Well Water Sampling	1
IC-CP-01-6289	Calibration of Meteorological Instrumentation	11
IC-FT-01-6289	Functional Test of Meteorological Instrumentation(supercedes IC-CP-01-6289)	0
IC-CP-03-0042	Calibration of AVS-28A Air Sampler	3

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
10-QUA-083	SARC Audit Report No. 63	December 28, 2010
CH-ST-MM-0002	Annual Radiological Effluent Report	April 2, 2009
CH-ST-MM-0002	Annual Radiological Effluent Report	March 19, 2010
CH-ST-MM-0002	Annual Radiological Effluent Report	March 10, 2011

CONDITION REPORTS

2011-10125	2011-10047	2011-10003	2011-09016	2011-08349
2011-07600	2011-05735	2011-05460	2011-04589	2011-03939
2012-02415	2012-02131	2012-01824	2012-01791	2011-10438
2012-03714	2012-03524	2012-03474	2012-03078	2012-02444
2012-04186	2012-04171	2012-03941	2012-03774	2012-03765
2012-05708	2012-05659	2012-05658	2012-05657	2012-05656
2012-05777	2012-05774	2012-05744	2012-05726	2012-05724

CALIBRATION AND MAINTENANCE RECORDS

NUMBER	TITLE	DATE
WO 00430019-01	Inspection and Cleaning of Rain Gauge	May 23, 2012
39328701	Attachment 9.2 – Test Record Package for IC-CP-01-6289	May 24, 2011
WO 432970-06	Weekly Environmental Radiation Air Sampler Operational Checks (performed on 6/20/2012)	June 20, 2012
M&TE# 04246	Attachment 9.1 Test Record Package for IC-CP-03-0042	May 3, 2010
M&TE# 04246	Attachment 9.1 Test Record Package for IC-CP-03-0042	March 14, 2011
M&TE# 04246	Attachment 9.1 Test Record Package for IC-CP-03-0042	November 9, 2011
M&TE# 04246	Attachment 9.1 Test Record Package for IC-CP-03-0042	June 21, 2012
M&TE# 04245	Attachment 9.1 Test Record Package for IC-CP-03-0042	February 5, 2010
M&TE# 04245	Attachment 9.1 Test Record Package for IC-CP-03-0042	April 12, 2011
M&TE# 04245	Attachment 9.1 Test Record Package for IC-CP-03-0042	May 10, 2012
	Minor Work Instruction: Cathodic Protection for Met Tower Anchors	

	Calibration Certificate for Climatronics Temp Probe Model No 1000093, T1 S/N 12, T2 S/N 887	May 16, 2012

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION/DATE
CH-ODCM-0001	Off-Site Dose Calculation Manual	20, 21
	2010 Radiological Environmental Operating Report	
	2011 Radiological Environmental Operating Report	
	2010 Annual Radiological Effluent Release Report	
	2011 Annual Radiological Effluent Release Report	
PLDBD-EV-70	Design Basis Document: Site Meteorology	6
12-001-0	Official Licensing Position: Commitment to NRC Regulatory Guide 1.23 and Requirement for Dew Point Instrumentation	April 23, 2012
	2011 Area TLD Results	March 5, 2012
WO 432970-06	Weekly Environmental Radiation Air Sampler Operational Checks	
	2010 Land Use Survey X/Q & D/Q values	
FC00082	361’ Tower Calculations	February 1976
D-100-036-760218	360’ T-36 Tower	February 18, 1976
D-100-036-76218-1	Base and Anchor Details	February 12, 1976

Section 2RS08: Radioactive Solid Waste Processing and Radioactive Material handling, Storage, and Transportation

PROCEDURES

NUMBER	TITLE	REVISION
FCSG-23	10 CFR 50.59 Resource Manual	8
NOD-QP-16	Updated Safety Analysis Report (USAR)	21
RP-204	Radiological Area Controls	61
RW-218	10 CFR 61 Classification	15
RW-300	Shipping Radwaste and Radioactive Materials	19
RW-AD-300	Process Control Program	0

CONDITION REPORTS

2012-05793	2012-05772	2012-05725	2012-05676	2012-04153
2012-03704	2011-08260	2011-03636	2010-06471	2010-06394
2010-06393				

RADIOACTIVE MATERIAL SHIPMENTS

NUMBER	TITLE	DATE
FCS NW 12-12	Equipment for Decontamination	June 21, 2012
FCS RW 12-04	Dry Active Waste for Processing	February 16, 2012
FCS RW 11-43	Dry Active Waste for Processing	December 11, 2011
FCS RW 11-42	Metal Oxide-Resin for Processing	December 6, 2011
FCS RW 11-23	Compacted Trash	May 18, 2011
FCS RW 10-20	Process Resins	April 15, 2010
FCS RW 10-03	Low Specific Activity Metal Oxides	February 18, 2010

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION/DATE
USAR Appendix G	Responses to 70 Criteria	21
EC No. 33099	Original Steam Generator Storage Facility	0
	2010 Annual Radiological Effluent Release Report	
	2011 Annual Radiological Effluent Release Report	
	Updated Safety Analysis Report – Chapter 11.2:Radioactive Waste and Radiation Protection and Monitoring	17
	Updated Safety Analysis Report – Chapter 11.3: Radiological Effluent Requirements	6
107C-865972-B	Updated Safety Analysis Report – Chapter 11.2: Radioactive Waste and Radiation Protection and Monitoring	May 22, 1985
107C-865212-B	Plant Drawing: Filter Housing, Carbon, VA-26A&B – SI Pump Room	June 24, 1970
CHRF1103	Outage Primary Chemistry Refresher	2012
	System Health Report for Radiation Effluent Monitors	2009-2012
	Annual Radiological Effluent Release Report	2010
	Annual Radiological Effluent Release Report	2011

Section 4OA4: IMC 0350 Inspection Activities

CONDITION REPORTS (CR)

2011-08957	2011-00319	2011-09793	2012-09767	2011-03009
2012-09296	2012-09722	2012-09771	2012-09725	2012-09865
2006-03665	2012-01630	2012-04342	2012-05015	2012-05035
2006-05248	2012-01765	2012-04344	2012-05018	2012-05037
2007-00334	2012-01799	2012-04345	2012-05019	2012-05038
2008-00409	2012-01868	2012-04346	2012-05020	2012-05088
2008-01769	2012-01868	2012-04347	2012-05021	2012-05294
2009-02306	2012-02652	2012-04348	2012-05022	2012-05615
2009-03476	2012-02652	2012-04349	2012-05023	2012-01655
2010-02929	2012-04114	2012-04350	2012-05024	2012-05967
2010-05140	2012-10391	2012-04351	2012-05025	2012-06076
2011-02804	2012-03986	2012-04352	2012-05026	2012-06715
2011-02976	2011-09459	2012-04353	2012-05027	2012-07860
2011-03004	2011-10162	2012-04354	2012-05028	2012-06700
2011-05400	2012-03351	2012-04355	2012-05029	2012-10612
2011-05414	2012-04307	2012-04356	2012-05030	2012-10382
2011-05514	2012-04320	2012-04357	2012-05031	2012-05032

2011-06621	2012-04321	2012-04988	2012-05033	2012-05034
2010-2387	2011-10302	2012-01021	2012-02142	2012-08911
2012-09265	2012-08614	2012-08653		

WORK ORDERS (WO)

422227	422228	427902	CWO181503

PROCEDURES

NUMBER	TITLE	REVISION
PED-SEI-31	Vendor Manual Control	5
SO-G-62	Control of Vendor Manuals	14
SO-R-2	Condition Reporting and Corrective Actions	52
FCS-65-2	Recovery Checklist Issue Closure	0
FCS-65-3	Restart Classification and Management of recovery Action Items under MC 0350 Restart Oversight	0
NP 95003 Admin C	Admin Controls for 95003 Work Scope for Station Recovery	1
PED-GEI-12	Design Engineering Review of Design Basis Document Revisions	3
PED-GEI-52	Preparation of Field Design Change Requests	2
PED-QP-13	Design Basis Document Control	7
SO-R-2	Condition reporting and Corrective Action	52
FCSG-65-2	FCS IMC 0350 Recovery Project. Issue Closure/ NRC Inspection Guideline	0
FCSG-65-3	Restart Classification and Management of Recovery Action items Under MC350 Restart Oversight	0
FCSG-65-2	Issue Closure / NRC Inspection Guideline	0
SO-R-2	Condition Reporting and Corrective Action	52

PROCEDURES

NUMBER	TITLE	REVISION
FCSG-24-1	Condition Report Initiation	0
FCSG-24-2	Evidence Quarantining	0
FCSG-24-3	Condition Report Screening	1
FCSG-24-4	Condition Report and Cause Evaluation	1
FCSG-24-5	Cause Evaluation Manual	0
FCSG-24-6	Corrective Action Implementation and Condition Report Closure	1
FCSG-24-7	Effectiveness Review of Corrective actions to Prevent Recurrence (CAPRs)	0
FCSG-24-8	Departmental Corrective action Review Board	1
FCSG-24-9	Station Corrective Action Review Board	1
FCSG-24-10	Corrective Action Program Trending	0
FCSG-24-12	Corrective Action Program Coordinator (CAPCO)	0
EPIP-TSC-2	Catastrophic Flooding Preparations	14
PE-RR-AE-1001	Flood Barrier and Sandbag Staging and Installation	16
FCSG-64	External Flooding of Site	2
SO-G-124	Flood Barrier Impairment	2
AOP-01	Acts of Nature	31
	Flood Demonstration: Post Exercise Report	8/6/2010

CALCULATIONS

NUMBER	TITLE	REVISION
EA-FC-91-084	Breaker Coordination Study	8
FC07397	Fort Calhoun Cycle 24 SBLOCA Analysis	0

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
ACA 2011-09276	Apparent Cause Analysis for Missed Vendor Manual	1

MISCELLANEOUS DOCUMENTS

CC-09315466-1	Certificate of Conformance for 1B4A Switchgear Refurbishment	0
EA-11-025	NRC M2 Contactor Final Significance Letter	18-Jul-11
EA53257	480V Bus 1B4A Repair/Replace	0
EC33464	Replace AK-50 480V Main & Bus-Tie Breakers with Molded Case Type Or Equivalent	0
ERPG-10CFR50.59-02	10CFR50.59 Document Review Task Familiarization Guide	0
ERPG-DNC/OPEVAL-01	Engineering Recovery Process Guide-Degraded/Nonconforming Conditions & Operability Evaluations	2
ERPG-ESL-01	Engineering Recovery Process Guide Equipment Service Life	0
ERPG-VM-01	Engineering Recovery Process Guide Vendor Manual	1
ERPG-VMOD-02	Vendor Modification Document Review Task Familiarization Guide	0
RCA 2011-0451	M2 Contactor Root Cause Analysis	Rev 1 and Rev 2
RCA 2012-03986	Organizational Effectiveness Root Cause Analysis	0
RSAC ESL	Equipment Service Life Recovery Scope and Approval Control	15-May-12
RSAC M2 Contactor	M2 Contactor Recovery Scope & Approval Control	8-Mar-12
RSAC Vendor Man	Vendor Manuals Recovery Scope & Approval Control	19-Apr-12

MISCELLANEOUS DOCUMENTS

SDBD-EE-201	AC Distribution	21 & 22
SEAD-36	Requalification Training	2012
TD N967.0040	Instruction Manual for NLI/Square D Masterpact Breaker/Craddle Assembly: SDS Part Number LGSB4	1
USAR Figure 8.1-1	Simplified one Line Diagram Plant Electrical System P&ID	142
USAR-14	Safety Analysis	130
USAR-5	Structure	130
USAR-7	Instrumentation and Control	130
USAR-8	Electrical Systems	130
WCAP-12476	Evaluation of LOCA during Modes 3&4 Operation for W NSSS	Nov-91
	1B3A Root Cause Status meeting minutes and agenda	7/19/12 & 8/16/12
	1B4A Root Cause Status meeting minutes and agenda	7/19/12 & 8/16/12
	Affected Documents Presentation	
	Collective Evaluation Streaming Analysis	7/18/2012
	Engineering Recovery Evaluation Area Familiarization Training	
	Engineering Recovery Overview Presentation	
	Fort Calhoun Station Key Performance Indicators Report	May-12

MISCELLANEOUS DOCUMENTS

Integrated Performance Improvement Plan	3
Purchase Order 00163495 Amendment 004	
Purchase Order 166406	---
Purchase Order Number 00164765	
Recovery Action Closure Verification Checklist, Action Item Number 2.3.1.13	20-Mar-12
Recovery Action Closure Verification Checklist, Action Item Number 2.3.1.14	19-Mar-12
Recovery Action Closure Verification Checklist, Action Item Number 2.3.1.15	20-Mar-12
Recovery Action Closure Verification Checklist, Action Item Number 2.3.1.16	20-Mar-12
Schneider Electric Micrologic Instruction Bulletin 48049-136-05	
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