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

December 31, 2012

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

Subject: FORT CALHOUN - NRC INTEGRATED INSPECTION REPORT NUMBER 05000285/2012011

Dear Mr. Cortopassi:

On November 18, 2012 the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. The enclosed inspection report documents the inspection results which were discussed on December 6, 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.

Four NRC identified and one self-revealing finding of very low safety significance (Green) were identified during this inspection. Four of these findings were determined to involve violations of NRC requirements.

If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at 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 the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michael Hay
Chief, Project Branch F
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report 05000285/2012011
w/Attachment: Supplemental Information

cc w/ encl: Electronic Distribution

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Inspection Reports/MidCycle and EOC Letters to the following:
ROPreports

ADAMS ACCESSION NUMBER: ML12366A158

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

SRI:DRP/F	RI:DRP/F	SPE:DRP/F	ROE:NRR/DIRS/IRIB	BC:DRP/F	
JCKirkland	JFWingebach	RWDeese	AKlett	MCHay	
/RA Via email/	/RA Via email/	/RA Via email/	/RA Via email/	/RA/	
12/28/12	12/28/12	12/28/12	12/27/12	12/31/12	

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2012011
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	October 1 through November 18, 2012
Inspectors:	J. Kirkland, Senior Resident Inspector J. Wingebach, Resident Inspector R. Deese, Senior Project Engineer F. Ramirez, Resident Inspector A. Klett, Reactor Operations Engineer W. Smith, Project Engineer J. Brand, Reactor Inspector D. Stearns, Health Physicist C. Alldredge, Health Physicist
Approved By:	Michael Hay, Chief, Project Branch F Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2012011; 10/01/2012 – 11/18/2012; Fort Calhoun Station, Integrated Resident and Regional Report; and Radiological Hazard and Exposure Control

The report covered a 6-week period of inspection by resident inspectors and an announced baseline inspection by a region-based inspector. Four Green non-cited violations and one green finding of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The cross-cutting aspect is determined using Inspection Manual Chapter 0310, "Components Within the Cross-Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The NRC identified a non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Procedures," for failing to follow a quality procedure. Specifically; PED-QP-13 "Design Basis Document Control," requires FCS to update and maintain their Design Bases Documents. The license has failed to maintain these design documents. Some examples include PLDBD-51 "Seismic Criteria" where the configuration of the Steam Generator supports were not accurately described, and PLDBD-ME-10 "Pipe Stress and Supports" where the piping design code classification for Main Steam is incorrect. The licensee entered the issue into its corrective action program for evaluation and review.

  The performance deficiency is more than minor because if left uncorrected it would have the potential to lead to a more significant safety concern. The finding was determined to affect the Initiating Events, Mitigation Systems, and Barrier Cornerstones using Inspection Manual Chapter 0609.04, "Initial Characterization of Findings." The finding was characterized as having very low safety significance (i.e., Green) using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At-Power," because all logic questions for the applicable cornerstones were answered in the negative. The finding is assigned a cross-cutting aspect in the area of Human Performance, in the component of Resources because the licensee failed to ensure that personnel, equipment, procedures, and other resources, specifically those necessary for complete, accurate and up-to-date design documentation, were available and adequate to assure nuclear safety. H.2(c) (Section 4OA4).

- Green: The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure that adequate equipment was available to measure river level locally to comply with an abnormal operating procedure. Specifically, the length of the weighted tape measure used to measure river level locally was inadequate to ensure that the entire range of river levels needed for operation of the plant would be covered. The licensee entered the issue into its corrective action program for evaluation and review.

The performance deficiency was determined to be more than minor because it is associated with the Mitigating Systems Cornerstone attribute of Equipment Performance and it adversely affects the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was screened as very low safety significance (Green) because the licensee maintained an adequate mitigation capability and it would not be characterized as a loss of control. The inspectors determined the finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee falied to thoroughly evaluate problems such that resolutions address the causes and extent of condition specifically associated with deficiencies involving the "Acts of Nature" procedural guidance (P.1(c)), (Section 4OA4).

- Green: The inspectors identified a finding of very low safety significance (Green) for the licensee's failure to generate a complete inspection list, with all the external flood protection features credited in the current licensing basis documents for flooding events, to comply with NRC endorsed NEI 12-07, "Guidelines for Performing Walkdowns of Plant Flood Protection Features." These walkdowns were being performed in response to a March 12, 2012, letter from the NRC to licensees, entitled, "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident." Specifically, the scoping list did not include several active components, which are an essential part of Fort Calhoun's design basis flood mitigation strategy. The licensee entered the issue into the corrective action program and revised the scoping list accordingly.

  The performance deficiency was determined to be more than minor because it is associated with the Mitigating Systems Cornerstone attribute of Protection Against External Factors (Flood Hazard) and it adversely affects the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, in addition to not scoping the sluice gates into the Flooding Features Walkdown List, fourteen additional active components would not have been scoped into the walkdown list. This would have prevented the licensee from identifying that preventive maintenance tasks needed to be created, and some active components that are an essential part of the flood mitigating strategy would not have been inspected and tested. The finding was screened as very low safety significance (Green) because the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding, or severe weather initiating event. The inspectors determined the finding had a cross-cutting aspect in the area of human performance because licensee personnel did not properly apply human error prevention techniques such as peer checking and proper documentation of activities (H.4(a)) (Section 4OA5).

Cornerstone: Barrier Integrity

- Green. The NRC identified a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, “Procedures,” for the failure to perform an adequate operability determination as required by FCS Procedure NOD-QP-31, “Operability Determination Process.” Specifically, the licensee’s operability determination for non-conforming containment internal structures failed to address that a section of the containment internal structures exceeded the allowable working stress criteria. The licensee entered this issue into its corrective action program for evaluation and review.

  Inspectors found that the failure to perform an adequate operability determination to specifically evaluate that the containment internal structures did not meet the design code of record was a performance deficiency. This violation is more than minor because it is associated with the design control attribute of the barrier integrity cornerstone and has the potential to adversely affect the cornerstone objective. The inspectors used Inspection Manual Chapter 0609, Appendix G “Shutdown Operations Significance Determination Process”, to determine that the issue screened as very low safety significance (green) because it did not require a quantitative assessment per Checklist 4. This violation was determined to have a crosscutting aspect in the area of human performance associated with decision making [H.1.b]. Specifically, the licensee did not use conservative assumptions in decision making and did not adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action (Section 1R15).

Cornerstone: Occupational Radiation Safety

- Green. Inspectors reviewed a self-revealing Green noncited violation of Technical Specification 5.8.1.a for the failure to follow procedure requirements related to radiation work permit requirements. Specifically, workers unexpectedly created a high radiation area when working with tri nuke filter hosing while on a radiation work permit that did not allow access into a high radiation area. Both workers received alarms on their dosimeters. The licensee entered the issue into its corrective action program for evaluation and review.

  The failure to follow a procedure was a performance deficiency. The finding was more than minor because it negatively impacted the Occupational Radiation Safety cornerstone’s attribute of program and process, in that not following the requirements of the radiation work permit led to workers’ unplanned, unintended dose. Using NRC Manual Chapter 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” the finding was determined to be of very low safety significance because: (1) it was not associated with as low as is reasonably achievable (ALARA) planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. The finding has a problem identification and resolution crosscutting component associated with operating experience because the licensee didn't implement operating experience through changes to station procedures. Specifically, there was operating experience which could have prevented the issue if it had been discussed at the pre-job brief [P.2.b] (2RS01).

B. **Licensee-Identified Violations**

None

REPORT DETAILS

Summary of Plant Status

The station remained in Mode 5 with the fuel in the spent fuel pool for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

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

- November 4, 2011, Fire Area 32, Compressor Area (Rooms 19, 50 through 53)
- November 4, 2012, Fire Area 34A, Electrical Penetration Area – Basement (Room 20)
- November 4, 2012, Fire Area 34B, Electrical Penetration Area – Ground and Intermediate Areas (Room 57)
- November 4, 2012, Fire Area 34C, Group 1 MCC Area (Room 57)

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

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

b. Findings

No findings were identified.

1R15 Operability Evaluations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following assessments:

- Review of the operability assessment for moving fuel from the reactor vessel to the spent fuel pool completed on August 29, 2012

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

These activities constitute completion of one operability evaluations inspection sample(s) as defined in Inspection Procedure 71111.15-05.

b. Findings

Introduction: The NRC identified a green non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V "Procedures", for the failure to perform an adequate operability determination as required by FCSProcedure NOD-QP-31, "Operability Determination Process." Specifically, the licensee's operability determination for non-conforming containment internal structures failed to adequately evalaute that a section of the containment internal structures exceeded the allowable working stress criteria.

Description: On May 22, 2012, OPPD generated a condition report (CR 2012-04392) to track and document the extent of condition of the design methods and validity of calculations for the containment interior structures. This condition report was generated because several issues with the design of containment internal structures were identified earlier in the year. Specifically, errors included: failure to meet working stress and no loss of function design criteria required by the USAR, multiple calculations of record, discrepancies between as-built design drawings and calculations, discrepancies in loading values for the same structures, unchecked load combinations, assumptions without justification, numerical errors, poor legibility, lacking calculations, etc. Some errors were simply poor documentation, but several errors are significantly non-conservative and call into question the ability of the structure to support its design basis function.

On August 29, 2012, OPPD completed an operability determination (CR 2012-11933-1) to evaluate the operability of the fuel transfer canal and fuel handling machine inside containment. The operability concern was generated to evaluate the removal of fuel from the reactor to the spent fuel pool. The operability determination concluded that the structure “is structurally sound and will continue to perform its Technical Specification 2.8 support function”; therefore it was operable but degraded. However, the operability determination also determined that the structure exceeded the allowable working stress criteria.

USAR Section 5.11 specifies that the 1963 edition of ACI-318 is the code of record for safety related concrete structures at Fort Calhoun Station (FCS). ACI-318 requires that the containment internal structures meet both working stress and no loss of function (ultimate strength) design criteria. NRC Inspection Manual Chapter Part 9900, section C.13 “Structural Requirements” states:

“Structures may be required to be operable by the TSs, or they may be related support functions for SSCs in the TSs... If a structure is degraded, the licensee should assess the structure’s capability of performing its specified function. As long as the identified degradation does not result in exceeding acceptance limits specified in applicable design codes and standards referenced in the design basis documents, the affected structure is either operable or functional.”

Fort Calhoun Station Procedure NOD-QP-31, “Operability Determination Process”, Revision 51, Step 4.1.3 J requires that “a positive determination of operability must be justified, including…a technical discussion of why the concern identified does not prevent the item from fulfilling its intended safety function(s). This should demonstrate that the item is not exceeding its design basis specified in the reference documents.” Contrary to NOD-QP-31, FCS concluded that the containment internal structures were operable despite the structure exceeding the working stress acceptance limits specified in the applicable design code (ACI-318) without a technical discussion.

The inspectors noted that the licensee is currently in the process of reconstituting the design basis of the containment internal structure and performing detailed analysis to assess both the working stress and no loss of function design criteria. These analysis are under review by the NRC and will be completed prior to restart of the facilty to ensure the plant is safe to operate.

<u>Analysis</u>: Inspectors found that the failure to perform an adequate operability determination and to identify the containment internal structures did not meet the design code of record was a performance deficiency. This violation is more than minor because it is associated with the design control attribute of the barrier integrity cornerstone and

has the potential to adversely affect the cornerstone objective. The inspectors used Inspection Manual Chapter 0609, Appendix G "Shutdown Operations Significance Determination Process", to determine that the issue screened as very low safety significance (Green) because it did not require a quantitative assessment per Checklist 4.

This violation was determined to have a crosscutting aspect in the area of human performance associated with decision making [H.1.b]. Specifically, the licensee did not use conservative assumptions in decision making and did not adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action. For example, the licensee did not address the containment internal structures exceeding the allowable working stress in its evaluation which is not consistent with the USAR or the applicable design code. Additionally, after discussions with engineering and operations personnel, the inspectors determined the operability determination is not a standalone document because it did not explicitly cite the origin of numerical values used in the calculation.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, requires that activities affecting quality be prescribed by procedures and be accomplished in accordance with those procedures. FCSProcedure NOD-QP-31, "Operability Determination Process", Revision 51, Step 4.1.3 J requires that "a positive determination of operability must be justified, including…a technical discussion of why the concern identified does not prevent the item from fulfilling its intended safety function(s). This should demonstrate that the item is not exceeding its design basis specified in the reference documents." Contrary to NOD-QP-31, FCS concluded that the containment internal structures were operable despite the structure exceeding the working stress acceptance limits specified in the applicable design code (ACI-318) without a technical discussion.

Since this violation was of very low safety significance and was documented in the licensee's corrective action program it is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012011-01, "Inadequate Operability Determination for Containment Internal Structures."

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS01 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

This area was inspected to: (1) review and assess licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities and the implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures, (2) verify the licensee is properly identifying and reporting Occupational Radiation Safety Cornerstone performance indicators, and (3) identify those performance deficiencies that were reportable as a performance indicator and which may have represented a substantial potential for overexposure of the worker.

The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed walkdowns of various portions of the plant, performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation reported by the licensee in the Occupational Radiation Safety Cornerstone

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

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

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

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

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

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

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 71124.01-05.

b. Findings

Introduction. Inspectors reviewed a self-revealing Green noncited violation of Technical Specification 5.8.1.a for the failure to follow procedure requirements related to radiation work permit requirements. Specifically, workers unexpectedly created a high radiation area while on a radiation work permit that did not allow access into a high radiation area. Both workers received alarms on their dosimeters.

Description. On September 13, 2012, work was performed in preparation for diving activities under Radiation Work Permit 11-1508 Task 5 on the 1036 foot level of containment in the reactor cavity, a posted radiation area. Task 5 authorized the workers for entry into radiation areas, contaminated areas, highly contaminated areas, and airborne radioactivity areas, but did not authorize them for entry into high radiation areas. The workers' electronic dosimetry had a dose rate alarm set point of 100 mrem/hr. During this work activity, workers changed the filter for the tri-nuke and brought the hose connection to the surface of the water to untangle the hoses. With the two hoses connected, dose rates were less than 100 mrem/hr at 30 cm, so the area was properly considered to be a radiation area. To untangle the hoses, the workers uncoupled the two hoses and pulled them apart above the surface of the water. At this point, both workers received an electronic dosimeter alarm. One worker received an alarm of 501mrem/hr and the other received an alarm of 232 mrem/hr. Uncoupling the hose above the water caused the area to become a high radiation area, which the workers were not authorized for entry into and were not briefed for according to their radiation work permit. Industry operating experience existed where contaminated tri-nuke hoses resulted in dose rate alarms when breached, but no operating experience was covered in the pre-job brief for this work activity.

The licensee placed the finding into the corrective action program as Condition Report CR 2012-13327.

Analysis. The failure to follow a procedure was a performance deficiency. The finding was more than minor because it negatively impacted the Occupational Radiation Safety cornerstone's attribute of program and process, in that not following the requirements of the radiation work permit led to workers' unplanned, unintended dose. Using NRC Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the finding was determined to be of very low safety significance because: (1) it was not associated with ALARA planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. The finding has a problem identification and resolution cross-cutting component associated with operating experience because the licensee didn't implement operating experience through changes to station procedures. Specifically, there was operating experience which could have prevented the issue if it had been discussed at the pre-job brief [P.2.b].

Enforcement. Technical Specification 5.8.1.a requires that written procedures be established, implemented and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Section 7.e, covers exposure controls, including access control to radiation areas including a radiation work permit system. Fort Calhoun Procedure SO-G-101 “Radiation Worker Practices,” step 4.4.4 B., states that personnel signed in on a radiation work permit shall adhere to the requirements and instructions listed on the radiation work permit. Contrary to the above, on September 13, 2012, workers did not adhere to the requirements of their radiation work permit. Specifically, workers unintentionally created a high radiation area in their work area by uncoupling two tri-nuke hoses above the surface of water, while they were only authorized for work in a Radiation Area. Since this violation was of very low safety significance and was documented in the licensee’s corrective action program as Condition Report CR 2012-13327, it is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012011-02, “Failure to Follow Radiation Work Permit Requirements”

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

This area was inspected to assess performance with respect to maintaining occupational individual and collective radiation exposures ALARA. The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee’s procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed licensee personnel and reviewed the following items:

- Site-specific ALARA procedures and collective exposure history, including the current 3-year rolling average, site-specific trends in collective exposures, and source-term measurements

- ALARA work activity evaluations/postjob reviews, exposure estimates, and exposure mitigation requirements

- The methodology for estimating work activity exposures, the intended dose outcome, the accuracy of dose rate and man-hour estimates, and intended versus actual work activity doses and the reasons for any inconsistencies

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

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

- Audits, self-assessments, and corrective action documents related to ALARA planning and controls 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 71124.02-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.5 Safety System Functional Failures (MS05)

a. Inspection Scope

The inspectors sampled licensee submittals for the safety system functional failures performance indicator for the period from the fourth quarter 2011 through the third quarter 2012. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73." The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports, and NRC integrated inspection reports for the period of October 2011 through September 2012 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of one safety system functional failures sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified. The inspectors did identify however that the licensee failed to report one event as a safety system functional failure. Licensee Event Report 2012-012, "Multiple Safety Injection Tanks Rendered Inoperable" describes events where the licensee routinely rendered several safety injection tanks inoperable. The licensee correctly noted in the licensee event report that this represents a safety system functional failure, but failed to include this in reporting the performance indicator. Since the performance indicator was already of white significance, there was no significance in the licensee's failure to report this failure.

.16 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors reviewed performance indicator data for the second quarter 2012 through the third quarter 2012. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed corrective action program records associated with high radiation area (greater than 1 rem/hr) and very high radiation area non-conformances. The inspectors reviewed radiological, controlled area exit transactions greater than 100 mrem. The inspectors also conducted walkdowns of high radiation areas (greater than 1 rem/hr) and very high radiation area entrances to determine the adequacy of the controls of these areas.

These activities constitute completion of the occupational exposure control effectiveness sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed performance indicator data for the second quarter 2012 through the third quarter 2012. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed the licensee's corrective action program records and selected individual annual or special reports to identify potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose.

These activities constitute completion of the radiological effluent technical specifications/offsite dose calculation manual radiological effluent occurrences sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

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

.1 (Closed) 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.

This licensee event report is closed. Revision 1 of this licensee event report was submitted on November 16, 2012.

.2 (Open) Licensee Event Report 05000285/2012-003-01: 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.

The apparent cause was identified to be inadequate use of vendor oversight when design information was transmitted to the vendor. The analysis also identified a contributing cause of inadequate review of the calculation provided by the vendor during the owner acceptance process. Procedural requirements to conduct peer reviews prior to transmitting design information to vendors and contractors preparing safety-related calculations have been incorporated into the governing procedures. Additional corrective actions will revise the deficient calculation and procedure.

.3 (Closed) 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 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 Fort Calhoun Station. 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.

This licensee event report is closed. Revision 2 of this licensee event report was submitted on October 27, 2012.

.4 (Open) Licensee Event Report 05000285/2012-004-02: Inadequate Analysis of Drift Affects Safety Related Equipment

While investigating industry operating experience, it was determined that FCS is subject to similar conditions where Static "0" Ring pressure switches with certain housing styles exhibit a setpoint shift when exposed to a change in temperature if the switch body is not vented. FCS pressure switches that provide signals for high containment pressure to the reactor protection system and engineered safeguards actuation circuitry may have this configuration. The impact of the potential drift was evaluated and it was initially determined that neither reactor protection system nor the engineered safeguard circuitry may actuate at the required containment pressure of 5 psig. A subsequent evaluation of actual data concluded that safety analysis limits were not exceeded. However, two Technical Specification limits were not protected by the calibration procedure nominal trip setpoint when applying the additional uncertainty.

The Apparent Cause was determined to be poor vendor documentation which led to Engineering personnel to improperly interpret and apply the information contained in the Static “O” Ring vendor manual. Corrective actions were initiated to remove the vent caps, revise the affected calculations to the temperature correction factor and drift. Additional actions to revise and re-perform surveillance testing were initiated.

.5 (Closed) Licensee Event Report 05000285/2012-007-00: Failure of Pressurizer Heater Sheath

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 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.

This licensee event report is closed. Revision 1 of this licensee event report was submitted on October 27, 2012.

.6 <u>(Open) Licensee Event Report 05000285/2012-007-01: Failure of Pressurizer Heater Sheath</u>

On May 9, 2010, during power operation, pressurizer heater number 26 failed on ground fault. There was no indication of a reactor coolant system barrier breach at that time. 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 heater sheath 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 was an axial crack showing some branching. The crack was 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.

The root cause of this failure is fabrication of the heater sheath during the manufacturing process induced high tensile residual stresses on the outer surface of the sheaths.

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

.7 <u>(Open) Licensee Event Report 05000285/2012-017-00: Containment Valve Actuators Design Temperature Ratings Below those Required for Design Basis Accidents</u>

While performing an extent of condition review associated with the adequacy of air operated equipment inside containment to withstand containment main steam line break (MSLB) and loss of coolant accident (LOCA) temperatures, it was discovered that valves HCV-238 (Reactor Coolant System (RCS) Loop 1a Charging Line Stop Valve), HCV-239 (RCS Loop 2a Charging Line Stop Valve), and HCV-240 (Pressurizer RC-4 Auxiliary Spray Inlet Valve) have nitrile based elastomers for the air filter regulator and actuator and may not be able to withstand Containment MSLB and LOCA temperatures. The design temperature limit for the nitrile elastomers used in the valves is 180°F which is acceptable for the normal operating conditions inside containment of 120°F. However, during the MSLB and LOCA accident the temperature inside containment is analyzed to reach 370°F. Since these valves have both open and close functions supported by an air accumulator, failure of the nitrile based elastomers could prevent the valves from fulfilling their intended safety function.

A cause analysis is in-process. When completed, this LER will be supplemented.

.8 (Open) Licensee Event Report 05000285/2012-018-00: Containment Air Cooling Units Operated Outside of Technical Specifications during Cycle 26

While performing NRC Inspection Manual Chapter 0350 checklist reviews, the recovery engineering team identified that the containment air cooling and filtering system was operated outside its design basis during cycle 26 resulting in FCS being in a condition prohibited by Technical Specifications during that operating cycle.

A cause analysis is in-process. When completed, this LER will be supplemented.

.9 (Open) Licensee Event Report 05000285/2012-019-00: Traveling Screen Sluice Gates Found with Dual Indication

On August 14, 2012, at approximately 2100 hours Central Daylight Time, Operations was cycling all six traveling screen sluice gates when it was identified that traveling screen sluice gate CW-14E motor was stopping on high torque and provided indication that the gate was approximately 8 inches open. Traveling screen sluice gate CW- 14C was also stopping on high torque and providing indication the gate was not fully closed. During a flooding event, these sluice gates are credited to fully close allowing control of the intake structure cell level with the raw water pumps. Cell level is maintained below elevation 1,007 foot, 6 inches. This is the point at which the raw water pump bay could become flooded causing a loss of raw water to the component cooling water heat exchangers. On August 25, 2012, divers removed the sediment and debris from all sluice gate bottoms returning the sluice gates' capability to be fully closed In the event of a design basis flood.

The apparent cause of the failure of the sluice gates to fully close was debris under the gates. A cause analysis is in process and when completed; this LER will be supplemented.

4OA4 IMC 0350 Inspection Activities (92702)

Inspectors continued with IMC 0350 inspection activities, which include follow-up on the restart checklist contained in Confirmatory Action Letter (CAL) 4-12-002 issued June 11, 2012. The purpose of these inspection activities is to assess the licensee's performance and progress in addressing its implementation and effectiveness of FCS' Integrated Performance Improvement Plan (IPIP), significant performance issues, weaknesses in programs and processes, and flood restoration activities.

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 October 1 through

November 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. These assessments are documented in section 4OA4.5.

.a Flooding Issue – Yellow Finding

Item 1.a is included in the restart checklist for the failure of FCS 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) Inspection Scope

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 continued to verify that corrective actions are adequate to address the root and contributing causes.

Additionally, during the inspection period the Corps of Engineers was significantly reducing upstream dam release rates resulting in river level at the FCS lowering. To address these low river level conditions the inspectors reviewed the stations processes and procedures for maintaining the plant in a safe condition for abnormally low river level conditions.

(2) Assessment

The inspectors' review focused mainly on the adequacy of procedures that are associated with mitigation strategies for low river level conditions. As a result of the various procedure walk-downs, the inspectors had observations associated with the ability to implement procedural guidance specifically related to measuring low river level conditions. This observation was provided to the licensee and was placed in the Corrective Action Program.

The inspectors started reviewing the basis for the number of hours that the licensee bases their low river level plans on to ensure that the technical foundation for that window of preparation time was adequate. These inspection activities are still being conducted and will be documented in future inspection reports.

(3) Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion III, “Design Control”, for the licensee’s failure to ensure that adequate equipment was available to measure river level locally to be able to comply with an abnormal operating procedure. Specifically, the length of the weighted tape measure used to measure river level locally was not enough to ensure that the entire range of river levels needed for operation of the plant would be covered. As a result, FCS would not have been able to comply with the steps of AOP-1, Section IV, “Low River Level,” for a river level lower than 982 feet.

Description: While conducting a walkdown of AOP-1, Section IV, “Low River Level,” the inspectors noted that step 7 of the procedure stated: “Verify that all of the following level indications agree within 3 inches: ERF point L1900, LI-1900 Circ Water Pump Cell and River Lever Indicator and Local indication from the Intake Structure Veranda.” The response not obtained of that step stated: “If a level indication does not agree within 3 inches then utilize local indication.” To obtain local indication for the river level at Fort Calhoun Station, operations personnel use a tape measure that is attached to a weight, commonly known as a plumb bob. The plumb bob is dropped from the Intake Structure veranda to the surface of the river water below and the level is obtained from the length of tape measured between those two points (veranda floor and water surface level).

The site elevation of the veranda is 1,007 feet. Following discussions with licensee personnel, the inspectors found that the plumb bob currently used by the site is 25 feet long. As a result, that tape measure can only measure the river level to 982 feet. Additionally, Fort Calhoun Technical Specification 2.16 “River Level,” has an action statement that requires that at 976 feet, 9 inches the plant be placed in cold shutdown. This action protects the safety-related raw water pumps from lower river levels that might not provide enough net positive suction head to operate. Consequently, the operators would not have been able to comply with the AOP-1, Section IV for any river levels lower than 982 feet.

Analysis: The inspectors determined that the failure to ensure that adequate equipment was available to measure river level locally to be able to comply with an abnormal operating procedure was a performance deficiency warranting further evaluation. Using the guidance in IMC 0612, “Power Reactor Inspection Reports,”

Appendix B, “Issue Screening,” the inspectors determined this finding affected the Mitigating Systems cornerstone. The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of Equipment Performance and it adversely affects the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” and conducted a Phase 1 characterization and initial screening. Using Phase 1 Table 3, “SDP Appendix Router,” the inspectors answered ‘yes’ to the following question: “Does the finding pertain to operations, and event, or a degraded condition while the plant was shut down?” As a result, the inspectors were directed to use IMC 0609 Appendix G, “Shutdown Operations Significance Determination Process.” Using Appendix G the inspectors determined that the finding did not need a quantitative assessment because the licensee maintained an adequate mitigation capability and it would not be characterized as a loss of control. Consequently, the finding screened as Green. The inspectors determined the finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee falied to thoroughly evaluate problems such that resolutions address the causes and extent of condition specifically associated with deficiencies involving the “Acts of Nature” procedural guidance (P.1(c)).

Enforcement: Criterion III of 10 CFR 50, Appendix B, “Design Control,” states in part that measures shall me established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of structures, systems and components.

Contrary to this requirement, as of November 2012, the licensee failed to ensure that the equipment to obtain river level locally was suitable for the applicable range of river levels that needed to be measured. As a result the licensee could not comply with AOP-1, Section IV, “Low River Level,” for river levels lower than 982 feet.

The licensee entered this issue into their CAP as CR 2012-17853. Corrective actions planned by the licensee include changing the weighted tape measure with one that is appropriate for the range of levels needed to be measured. Because the licensee has entered the issue into their corrective action program and the finding is of very low safety significance, this violation of 10 CFR 50, Appendix B, Criterion III, is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2012011-03, “Failure to Ensure that Adequate Equipment was Available to Measure River Level Locally to be Able to Comply with an Abnormal Operating Procedure.”

.c Electrical Bus Modification and Maintenance – Red Finding

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 resulted in a fire on June 7, 2011, in the safety-related 480 volt (V) electrical switchgear. These deficiencies resulted in a finding having high safety significance.

(1) Inspection Scope

During the inspection period, the NRC continued to assess the status of 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, which were conducted November 7 - 16, 2012, included a walk-down of the remains of the fire-damaged breaker, a tour of the switchgear rooms, and interviews and discussions with licensee staff. Inspectors also interviewed FCS operators who were part of the station's fire brigade that responded to the fire event. The in-office activities, which were conducted at the inspectors' normal duty stations, consisted of reviews of documents associated with the recovery efforts, conditions reports, root cause analyses, scoping procedures, calculations, and drawings. Inspectors also reviewed the breaker vendor's (NLI and Square-D) independent root cause analysis of the fire event.

On November 13 and 15, 2012, the inspectors observed electricians and engineers perform inspections (visual and boroscope inspections) and alignment checks on 480 V AC breakers 1B3B-1B3B and 1B4B-1B4B per WO-450346-01. In addition, the inspectors performed an independent inspection of the associated PT wire and bus stabs alignment and cradle finger cluster engagement with the bus stabs.

(2) Assessment

The licensee's closure package for the fire event was in progress as of the end of the inspection period. The licensee stated that the closure package will be ready for NRC inspection in January 2013. The licensee performed an apparent cause evaluation for Violation 2012007-02, which was related to the stations' fire brigade response to the fire event. The licensee stated that the closure package for addressing that violation should also be ready for NRC inspection in January 2013.

IR 05000285/2012005 documented several conditions that either contributed to the initiation of the fire event or the unexpected electrical distribution system response. Inspectors reviewed the root cause analysis of the breaker fire prepared by Nuclear Logistics Inc. and Square-D. The vendors root cause concluded that the insulation of the PVC jacketed control wires used between the three main buses and the three potential transformers degraded, causing a phase-to-phase short circuit between two of the wires, which developed into an arcing fault. OPPD staff indicated during discussions with inspectors that this event was unlikely because the gauge of the control wires would result in an open circuit if overheated; however, OPPD staff was visually inspecting the control wires for signs of insulation degradation during the 480-V bus refurbishments. Inspectors will continue to review OPPD's corrective actions related to this issue. Inspectors are continuing to follow up on the issues of concern regarding Breaker 1A4-10's trip setpoints and the 480 V bus separation design.

(3) Findings

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

.2 <u>Flood Restoration and Adequacy of Structures, Systems, and Components</u>

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 FCS are in appropriate condition to support safe restart and continued safe plant operation. Section 2 reviews will also include an assessment of how the licensee appropriately addressed the NRC Inspection Procedure 95003 key attributes.

.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, FCS issued Revision 1 to the "FCS 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 CAL 4-11-003, listing 235 items described in the FCS 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) <u>CAL Action Item 2.2.1.2</u>

i. <u>Inspection Scope</u>

The purpose of Action Item 2.2.1.2 was to assess the effects of the flood on the Auxiliary Feedwater (AFW) System and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors queued condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The AFW System is provided for storage, pumping and delivery of makeup water to the steam generators in order to remove decay heat if the Main Feedwater System is not available. The AFW System consists of one emergency feedwater storage tank; one motor-driven and one turbine-driven auxiliary feedwater pump; one non-safety-related, diesel-driven auxiliary feedwater pump; one non-safety-related diesel fuel oil transfer pump and day tank; non-safety-related fuel oil piping and valves; remotely operated flow control valves; interconnecting piping to the Main Feedwater System and piping to the auxiliary feedwater nozzles in the steam generators.

The inspectors identified that no temporary modifications had been installed to combat the flood, however, the inspectors identified two preventive maintenance tasks that were deferred.

The diesel driven auxiliary feedwater pump, FW-54, is required to be started and ran once per month. A monthly run of the pump was completed on June 8, 2011. Due to the flood, the Condensate Storage Tank Isolation Valve, FW-684, was underwater and thus closed. With this valve closed, there was no suction source for FW-54, and the monthly runs were not performed in July or August. A subsequent monthly run was successfully completed on September 20, 2011.

The corrective action search yielded no condition reports written related to the flood or flood damage. The aforementioned valve FW-684 is associated with the Main Feed System and will be addressed in the evaluation of that system. The independent walkdown performed by the inspectors identified no adverse conditions to the AFW System and its individual components.

This activity constitutes completion of Action Item 2.2.1.2 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the AFW System. A detailed evaluation of the health of the AFW System will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.2 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(2) CAL Action Item 2.2.1.3

i. Inspection Scope

The purpose of Action Item 2.2.1.3 was to assess the effects of the flood on the Auxiliary Instrumentation System (AIS) and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The AIS consists of the Emergency Response Facilities Computer System (ERFCS), the Safety Parameter Display System (SPDS) and the Distributed Control System (DCS). The ERFCS acts as a computing platform for extensive plant-specific and emergency response requirements, the SPDS provides numerous channels of level, flow and pressure data to the real-time database for processing and display, and the DCS acts as the plant computer. The AIS is not safety related, however, is used to assist control room personnel in evaluating the safety status of Fort Calhoun Station.

The licensee temporarily relocated the ERF Host, PC-80B on June 4, 2011 in case the Technical Support Center was flooded. The inspectors verified that PC-80B was returned to the Technical Support Center and returned to service on September 6, 2011. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the AIS and its individual components.

This activity constitutes completion of Action Item 2.2.1.3 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the AIS. A detailed evaluation of the health of the AIS will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.21 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(3) CAL Action Item 2.2.1.4

i. Inspection Scope

The purpose of Action Item 2.2.1.4 was to assess the effects of the flood on the Control Rod Drive (CRD) System and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and September 30, 2012. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The CRD System provides a means to position the control rods for reactivity control during reactor startup, shutdown, and power operation. The control rods act in conjunction with the Reactor Protective System to provide a means for rapid reactor shutdown when limiting conditions are reached.

The inspectors identified no temporary modifications in place and no preventive or corrective maintenance were deferred because of the flooding. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the CRD System and its individual components.

This activity constitutes completion of Action Item 2.2.1.4 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the CRD System. A detailed evaluation of the health of the CRD System will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.9 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(4) CAL Action Item 2.2.1.5

i. Inspection Scope

The purpose of Action Item 2.2.1.5 was to assess the effects of the flood on the Chemical and Volume Control System (CVCS) and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The CVCS serves four major purposes. The purposes served by the system are: control of the Reactor Coolant System (RCS) volume/inventory, as indicated by the water level in the pressurizer; control of RCS chemistry to minimize corrosion and remove fission products; control of RCS purity to minimize the amount of activated corrosion products; and control of reactor reactivity during operation and refueling activities.

The inspectors identified no temporary modifications in place and no preventive or corrective maintenance were deferred because of the flooding. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the CVCS and its individual components.

This activity constitutes completion of Action Item 2.2.1.5 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the CVCS. A detailed evaluation of the health of the CVCS will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.5 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(5) <u>CAL Action Item 2.2.1.7</u>

i. <u>Inspection Scope</u>

The purpose of Action Item 2.2.1.7 was to assess the effects of the flood on the Emergency Core Cooling System (ECCS) and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The ECCS is required to provide emergency core cooling following a loss of primary or secondary coolant. Portions of the ECCS equipment are used to provide shutdown cooling. Auxiliary functions of the ECCS equipment include: fill and drain the refueling cavity; provide a backup cooling system for the Spent Fuel Pool Cooling System; provide a means of cooling containment spray water following a Recirculation Actuation Signal (RAS); and provide a means to fill and drain the safety injection tanks. The ECCS also provides water for initial fill and flushing of the reactor coolant pump mechanical seals.

The inspectors identified no temporary modifications in place and no preventive or corrective maintenance were deferred because of the flooding. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the ECCS and its individual components.

The inspectors analyzed pump performance data for the Low Pressure Safety Injection (LPSI) Pumps. These pumps are not normally operated for extended periods of time, but due to the extended shutdown, shutdown cooling has been in service since June 2011. The inspectors compared pump performance data for pumps SI-1A and SI-1B from 2011 and 2012 and noted no abnormalities.

This activity constitutes completion of Action Item 2.2.1.7 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the ECCS System. A detailed evaluation of the health of the ECCS will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.6 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(6) CAL Action Item 2.2.1.8

i. Inspection Scope

The purpose of Action Item 2.2.1.8 was to assess the effects of the flood on the Emergency Diesel Generator (EDG) System and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The emergency diesel generators are designed to furnish a reliable source of 4160V AC power for safe plant shutdown and operation of engineered safeguards when the normal sources of off-site power are lost. The diesel generators are normally aligned in a standby mode ready to automatically start, come up to rated speed and voltage, and energize the engineered safeguard buses when required. Each emergency diesel engine is supported by a dedicated Starting Air System, Scavenging Air System, Jacket Water System, Lube Oil System, Fuel Oil System, and required instrumentation and control.

Two temporary modifications were in place during the flooding. The first was to install a portable filtering device to allow for manual filtering of diesel fuel in the fuel oil storage tank. The portable filter, which was performed after the flood waters receded, was installed on September 14, 2011, and the inspectors verified it was removed September 15, 2011.

The second temporary modification involved installing an extension on the fill pipe for the diesel fuel oil storage tank, FO-1. With river levels expected to exceed 1,004 feet mean sea level (msl), the licensee installed an extension on the fill pipe such that the top of the fill pipe would be at an elevation of 1,014 feet msl. This extension was installed on June 11, 2011 and the inspectors verified it was removed on September 8, 2011.

The inspectors verified that no preventive or corrective maintenance were deferred during the flooding. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the EDG System and its individual components.

This activity constitutes completion of Action Item 2.2.1.8 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the EDG System. A detailed evaluation of the health of the EDG System will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.13 of the FCS Restart Checklist Basis Document. In addition, a comprehensive review to evaluate and verify the capability of the EDG System to fulfill its intended safety functions as defined by the licensing and design basis and identify broad-based safety, organizational, and performance issues will be conducted and documented in accordance with section 2.b.2 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(7) CAL Action Item 2.2.1.10

i. Inspection Scope

The purpose of Action Item 2.2.1.10 was to assess the effects of the flood on the Engineered Safeguards Features (ESF) System and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The ESF System provides for coordinated automatic actuation of systems which provide safety injection, containment isolation, containment spray, containment atmosphere cooling and filtering, containment ventilation isolation, auxiliary feedwater actuation, and steam generator isolation. The system includes control devices and circuits for automatic initiation, control, supervision, and testing. Secondary protection systems provide emergency boration, main steam isolation, and safety injection room and spent regenerant tank room full ventilation.

The inspectors identified no temporary modifications in place and no preventive or corrective maintenance were deferred because of the flooding. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the ESF System and its individual components.

This activity constitutes completion of Action Item 2.2.1.10 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the ESF System. A detailed evaluation of the health of the ESF System will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.23 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(8) CAL Action Item 2.2.1.11

i. Inspection Scope

The purpose of Action Item 2.2.1.11 was to assess the effects of the flood on the Fuel Handling System (FHS) and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The purpose of the FHS is to provide fuel handling equipment for the safe handling and movement of fuel assemblies including receipt of unirradiated fuel assemblies, placement in the reactor vessel during refueling, removal of spent fuel assemblies from the reactor vessel, underwater storage to remove decay heat, and transfer to spent fuel casks for shipment off site. The FHS is required to: minimize the potential for mishandling and reduce the chance of damage to the fuel assemblies and reactor components; provide maximum safety for personnel; and provide efficient fuel movement to minimize outage periods.

The inspectors identified no temporary modifications in place and no preventive or corrective maintenance were deferred because of the flooding. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the FHS and its individual components.

This activity constitutes completion of Action Item 2.2.1.11 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the FHS. A detailed evaluation of the health of the FHS will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.25 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(9) CAL Action Item 2.2.1.13

i. Inspection Scope

The purpose of Action Item 2.2.1.13 was to assess the effects of the flood on the Hoisting Equipment System (HES) and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The purpose of the HES is to provide hoists, cranes, winches, and other lifting devices necessary to move heavy loads throughout the different buildings and locations throughout the plant.

The inspectors identified no temporary modifications in place and no corrective maintenance was deferred because of the flooding. One preventive maintenance task was deferred due to the flood. The Trash Rack Sluice Air Hoist, HE-6, located in the intake structure was scheduled to have the hoist cleaned, inspected and lubricated by October, 2011. Due to the flood, this task was completed on October 25, 2011.

The corrective action search yielded three condition reports written related to the flood on the HES. The first involved the containment auxiliary crane, HE-44. The main power panel, MPP-23, is located outside just south of the equipment hatch. That panel was wetted during the flooding event. An internal inspection and cleaning of MPP-23 was performed on November 8, 2011.

The second involved the three job cranes located in the Fabrication Shop. These cranes had the pedestals submerged about 2-3 feet. The cranes were inspected and load tested on October 24, 2011.

The final condition report involved Hoist HE-6. The wall penetrations of the hoist were sealed with expanding foam prior to the flood. The foam was removed on October 18, 2011.

The independent walkdown performed by the inspectors identified no adverse conditions to the HES and its individual components.

This activity constitutes completion of Action Item 2.2.1.13 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the HES. A detailed evaluation of the health of the HES will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.7 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(10) CAL Action Item 2.2.1.14

i. Inspection Scope

The purpose of Action Item 2.2.1.14 was to assess the effects of the flood on the Instrument Air System (IAS) and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System. Though the action item states only the Instrument Air System, the inspectors assessed the Service Air System as well.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The Service and Instrument Air System provides compressed air to two separate headers. The service air header supplies air for sparging of the monitor tanks, operation of portable pneumatic tools, blowdown of the vacuum priming line drain tanks and other minor loads. The instrument air header supplies dry air for pneumatic instruments and controls, air-operated dampers, and air operated valves.

The inspectors identified no temporary modifications in place and no preventive or corrective maintenance were deferred because of the flooding. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the Service and Instrument Air Systems and their individual components.

This activity constitutes completion of Action Item 2.2.1.13 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the Service and Instrument Air Systems. A detailed evaluation of the health of the Service and Instrument Air Systems will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.19 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(11) CAL Action Item 2.2.1.15

i. Inspection Scope

The purpose of Action Item 2.2.1.14 was to assess the effects of the flood on the Main Feedwater (MFW) System and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The Feedwater and Condensate System transfers condensate from the main condenser hotwells in the turbine building to the steam generators in the containment, while improving system efficiency by increasing the temperature of the feedwater. The Heater Vents and Drains System provides a means of removing the condensed extraction steam and noncondensable gases from the shells of the feedwater heaters to maximize heater efficiency. During normal operation, the Feedwater System provides an adequate supply of heated feedwater to the secondary sides of the steam generators. The Feedwater Regulating System functions to control the steam generators at a programmed level by regulating the feedwater flow to the steam generators based on steam generator level and steam flow.

The inspectors identified no temporary modifications in place and no preventive or corrective maintenance were deferred because of the flooding.

The corrective action search yielded two condition reports related to the flood. These condition reports described mud and debris around the Condensate Storage Tank, DW-48, and its valves. The tank isolation valve, FW-684, and drain valve, FW-685, were cleaned of sand and mud and cycled open and closed three times. The tank level transmitter, LT-1191, was found wet and allowed to dry. The tank itself still had mud and debris near the bottom of the tank making early leak detection difficult, and that was cleaned away.

In addition, the contents of DW-48 were sampled during and after the flood to identify if there was any tank leakage. The inspectors compared the chemistry results and verified that the tank contents were within specification.

The independent walkdown performed by the inspectors identified no adverse conditions to the MFW System and its individual components. The inspectors also verified that the issues noted above had been satisfactorily completed.

This activity constitutes completion of Action Item 2.2.1.15 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the MFW System. A detailed evaluation of the health of the MFW System will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.15 of the FCS Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(12) CAL Action Item 2.2.1.17

i. Inspection Scope

The purpose of Action Item 2.2.1.17 was to assess the effects of the flood on the Radiation Monitoring System (RMS) and identify actions to restore the system. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The inspectors independently reviewed the system to identify if there were any temporary modifications in place as a result of the flood, if there were any outstanding preventive or corrective maintenance activities that had been deferred due to the flood, and reviewed condition reports to determine if there were any deficiencies noted due to the flood. The inspectors reviewed condition reports that were related to flooding, written between April 1, 2011 and December 31, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions related to flooding. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The Radiation Monitoring System (RMS) consists of permanently installed monitors that are divided into two general categories: the Process Radiation Monitoring System provides surveillance of plant effluent and critical process streams; and the Area Radiation Monitoring System provides surveillance of personnel exposure levels in hazardous and potentially hazardous plant areas.

The inspectors identified no temporary modifications in place and no preventive or corrective maintenance were deferred because of the flooding. The corrective action search yielded no condition reports written related to the flood or flood damage. The independent walkdown performed by the inspectors identified no adverse conditions to the RMS and its individual components.

This activity constitutes completion of Action Item 2.2.1.10 as described in CAL 4-12-002. It should be noted that the purpose of this action item was to assess the effects of the flood on the RMS. A detailed evaluation of the health of the RMS will be conducted prior to plant startup. This evaluation will be conducted and documented in accordance with section 2.b.1.24 of the FCS Restart Checklist Basis Document.

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.

.b Equipment Design Qualifications

This item of the Restart Checklist verifies that plant components are maintained within their licensing and design basis. 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. FCS will perform reviews to identify conditions where a non-safety-related component or subcomponent was improperly used in a safety-related application. The NRC assesses the licensee's equipment design qualifications review for inconsistent quality classifications. Additionally, the NRC assesses the licensee's review of the use of non-safety-related parts in safety-related applications. This will ensure proper design attributes have been incorporated and implemented.

(1) Inspection Scope

NRC inspectors reviewed the licensee's procedure and scope of work for assessing the safety-related parts program. Inspectors also interviewed station personnel and contractors that performed the reviews.

(2) Assessment

The licensee's closure package for the review of the safety-related parts program was in progress as of the end of the inspection period. The licensee stated that the closure package will be ready for NRC inspection in mid-January 2013. This date is subject to change.

The licensee's review consisted of identifying work orders (WOs) from 2007 to 2012 that involved the use of non-safety related parts in work tasks on safety-related systems, structures, or components (SSCs). The licensee increased its scope of reviewed WOs from approximately 2100 to 4300 after identifying an additional population of WOs that involved safety-related applications. The licensee generated approximately 30 condition reports related to non-safety related parts being used in safety-related applications. The licensee also identified issues related to how various parts or materials being stored in a warehouse were categorized during its transition to a new software system in 1998. This review was ongoing as of the end of the inspection period.

(3) Findings

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

.ii High Energy Line Break (HELB) Program and Equipment Qualifications (EQ)

Industry experience with extended power up-rates (a method some plants use to produce more power from the same reactor) highlighted potential problems associated with high energy line break effects. In preparations for a postponed extended power up-rate, FCS reviewed high energy line break calculations. FCS found that it was lacking adequate documentation and calculations for high energy line break effects in some areas. The NRC will assess and inspect the high energy line break analyses and documents to ensure the plant is within their license and design basis for high energy line break effects. The NRC will also inspect the licensee's qualifications and documentation to certify equipment for harsh environments. These equipment qualifications are required by regulations (e.g., 10 CFR 50.49).

(1) Inspection Scope

NRC inspectors reviewed the licensee's progress toward reconstitution of their High Energy Line Break program and Electrical Equipment Qualification (EEQ) program. The inspectors reviewed procedures, calculations, vendor documents and corrective action documents. Inspectors also interviewed station personnel that performed the reviews. Inspectors reviewed testing documents for containment penetration feed-throughs having Teflon insulation and sealing materials under the licensee's EEQ program.

(2) Assessment

The licensee's closure package for the review of its HELB and EQ programs was in progress as of the end of the inspection period. The licensee stated that the closure package will be ready for NRC inspection in mid-January 2013. This date is subject to change.

The licensee is reconstituting its EQ program because a 2007 self-assessment revealed deficiencies in system health reports and that the design basis was not well-tracked. As of the end of the inspection period, the licensee was about 75% through reassessing its harsh environment files and EQ binders to demonstrate qualification. The licensee stated that its closure package will contain the updated EQ binders, equipment walk-down lists, and WOs for modifications to minimize harsh environments or relocate equipment or to qualify components for harsh environments.

(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 restart checklist 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 will inspect to provide assurance that changes have been appropriately implemented.

(1) Inspection Scope

.i Vendor Modification Control

NRC inspections indicated that several vendor modification packages did not ensure critical characteristics were identified and properly addressed. To address this issue, FCS will review work performed by vendors. The NRC will evaluate the effectiveness of the vendor program to ensure adequate oversight of vendor work. NRC inspectors interviewed station personnel and contractors that performed the licensee's reviews of the vendor modifications.

.ii 10 CFR 50.59 Screening and Safety Evaluations

NRC inspections indicated that several changes to the facility were not properly screened or evaluated per the requirements 10 CFR 50.59. Plant and procedure modifications will be reviewed to determine if modifications required a 10 CFR 50.59 review. The assessment of Design Changes/Modifications will take into account the key attributes of Inspection Procedure 95003 (Sections 02.03 and 03.03). The NRC will evaluate the effectiveness of the licensee's 10 CFR 50.59 process to ensure proper treatment changes to the facility. NRC inspectors interviewed station personnel and contractors that performed the reviews of 50.59 documents.

(2) Assessment

The licensee's closure packages for the reviews of its 50.59 documents and vendor-prepared modification packages were in progress as of the end of the inspection period. The licensee stated that the vendor modifications and the 10 CFR 50.59 closure packages should be ready for NRC inspection in January 2013.

(3) Findings

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

.5 <u>Assessment of NRC Inspection Procedure 95003 Key Attributes</u>

Section 5 of the Restart Checklist is provided to assess the key attributes of NRC Inspection Procedure 95003. Performing Inspection Procedure 95003 will provide the NRC with supplemental information regarding licensee performance, as necessary to determine the breadth and depth of safety, organizational, and programmatic issues. While the procedure does allow for focus to be applied to areas where performance issues have been previously identified, the procedure does require that some sample reviews be performed for all key attributes of the affected strategic performance areas. The key attributes are listed as separate subsections below. It is intended that the activities in these subsections be conducted in conjunction with reviews and inspections for Sections 1 – 4, rather than a stand-alone review. The NRC will perform a detailed review of the auxiliary feedwater system as part of the Inspection Procedure 95003 assessment.

.a Design

<u>Engineering Design/Configuration Control Finding</u>

(1) Inspection Scope

The site performed an integrated assessment and identified fifteen Fundamental Performance Deficiencies that resulted in the overall performance decline at the station. One of the deficiencies identified was "Engineering Design/Configuration Control." Examples in this area included changes to plant configuration and design and licensing basis are not effectively analyzed, controlled, and implemented; incomplete documentation and poorly written justifications in modification packages; and evaluations of fit, form, and function have been inadequate.

The NRC will evaluate the thoroughness of the licensee's "Engineering Design/Configuration Control" assessment, adequacy of extent of condition and extent of causal analysis, and adequacy of associated corrective actions. As part of this inspection activity the inspectors identified the following finding.

(2) Finding

Introduction. The NRC identified a green non-cited violation of 10 CFR Part 50 Appendix B, Criterion V, "Procedures," for failing to follow a quality procedure. Specifically; PED-QP-13 "Design Basis Document Control," requires FCS to update and maintain their Design Bases Documents.

Description. The original Final Safety Analysis Report (FSAR) for FCS was principally prepared using an Atomic Energy Commission (AEC) document titled, "The Guide to the Organization and Contents of Safety Analysis Reports," June 1966. The rigid formality later imposed by the NRC's Standard Review Plan, Regulatory Guide 1.70 which specifies the standard Format and Contents of the Safety Analysis Report for Nuclear Power Plants, were not yet in place within the Commission in the late 60's and early 70's when FCS was undergoing licensing review.

The 1966 Guide laid out a pattern for the presentation of information based upon the following sequence:

a. Identification of the principal criteria for design of the facility and the design bases for those major systems and components significant to safety.
b. Description of how it was intended that the plant be built and operated to satisfy the principal criteria and design basis.
c. Systematic safety analysis and evaluation of the design that showed plant performance objectives could be achieved and safety assured.

It encouraged the use of a systematic and logical presentation of information associated with the evaluation of individual safety aspects of the particular plant.

In November 1976, the NRC published a proposed rule in the federal Register to require the holder of an operating license to provide the Commission periodically revised pages of its FSAR. The rule became effective on July 22, 1980 as 50.71(e) to 10 CFR Part 50. The purpose of the rule was to provide an updated reference document to be used in recurring safety analyses performed by Fort Calhoun Station, the Commission, and other interested parties. The rule did not impose a particular format for the update. The degree of detail to be maintained in the updated FSAR was to be at least the same as originally provided. A further understanding of the acceptable level of detail is given in the definition of Design Bases, Design Evaluation, and Safety Analysis as used in the original FSAR and stated in the 1966 Guide.

Several NRC inspections of FCS in 1985 highlighted several significant weaknesses. Specifically, NRC Inspection Report 50-285/85-22 states, "There appear to be several significant weaknesses which were identified in your design control processes. One of them was your failure to obtain, maintain, and use design basis information to assure that the original design margins are not unintentionally abrogated. We are also concerned that post-modification testing procedures were inadequate to confirm that the physical modifications fulfill the functional design requirements of the system or component. In general it was determined that the accessibility and retrievability of the original design specifications and some design basis information appeared to be a significant obstacle. There was also noted a strong over reliance on the USAR for such information. It is important to note that the lack of Design Basis Records had been identified by FCS as a generic concern prior to it being noted by the NRC. FCS had attempted to locate original architect/engineer design records on specific issues.

In response to NRC Violations from issues noted in NRC Inspection Reports 50-285/85-22 and 50-285/85-29 FCS docketed to the NRC a Corrective Action Implementation Plan. One item in this plan included the Reconstitution of Design bases. Specifically, "To locate and organize design bases records in such a way that a set of system oriented design bases documents (DBD) can be generated… These DBDs will be prepared to reflect the current design condition of the plant, combined with an historical perspective of the justification for the current plant configuration or generic subject area. The DBDs will be controlled documents to be updated as plant configuration or issues change. The primary purpose of the DBDs will be to evaluate the impact of modifications and changes in operating procedures, to support safety evaluations, and to determine the impact of new regulations or regulatory concerns" (LIC-87-691).

Based on NRC concerns and an independent assessment FCS had performed in 1988 on all its nuclear related activities FCS developed and docketed (LIC-88-1094) the Safety Enhancement Program (SEP). The purpose of the SEP was to consolidate the concerns that led to FCS being placed on the list of plants requiring additional NRC attention into a corrective action program leading to excellence. Item Number 4 of the SEP was to develop the DBDs. This item would constitute the Design Bases reconstitution and verification. FCS later committed item Number 4 of the SEP to the NRC in LIC-89-1006 with the objective to maintain the plant and system level DBDs for safety systems for the life of the plant.

PED-QP-13, Design Basis Document Control is the quality procedure FCS uses to control and maintain the DBDs. Section 4.2.1 states, "Once issued, the DBDs are high level design documents for the Fort Calhoun Station. These documents shall be used as the requirement source for developing configuration changes and as a reference for other activities including operations, testing, licensing, and training." Section 4.8 states," DBDs are lifetime QA Records as defined in the FCSQA Plan Section 3.4."

Based on condition reports and discussions with plant engineers and managers the NRC has determined that FCS has not been updating and maintaining the DBDs as required by PED-QP-13. Many condition reports indicate that the DBDs contain inaccurate, inadequate, or otherwise missing information. The NRC has become aware of a concern among FCS personnel regarding the historical quality of the DBDs in general.

FCS Root Cause Analysis (RCA) of Engineering Design/Configuration Control (2012-08125), recently determined, "There are known issues with the quality of DBDs and inconsistent guidance on how to use them…There is no formal guidance or process to locate and retrieve all design and licensing basis requirements associated with an engineering activity… The barrier of reliable design and licensing basis documents is failed." The RCA did not identify that FCS had realized they were violating a quality procedure by not updating and maintaining the DBDs. The RCA did not demonstrate knowledge of the historical perspective of the DBDs or the significance of inaccurate, inadequate, or otherwise missing information. Because of the above mentioned information the finding is NRC identified.

As previously discussed, the license has failed to maintain design documents. Some examples include PLDBD-51 "Seismic Criteria" where the configuration of the Steam Generator supports were not accurately described (CR2012-16416), and PLDBD-ME-10 "Pipe Stress and Supports" where the piping design code classification for Main Steam is incorrect (CR2012-00490). Currently these failures to update the design basis documents appear administrative in nature. Given the importance of maintaining these documents, since they provide an integral function used in FCS processes, the NRC will continue to focus inspection activities in the area of design control.

Analysis. Failing to follow a quality procedure is a performance deficiency. The performance deficiency is more than minor because if left uncorrected it would have the potential to lead to a more significant safety concern. The finding was determined to affect the Initiating Events, Mitigation Systems, and Barrier Cornerstones using Inspection Manual Chapter 0609.04, "Initial Characterization of Findings." The finding was characterized as having very low safety significance (i.e., Green) using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) For Findings At-Power," because all logic questions for the applicable cornerstones were answered in the negative. Because FCS continues to not maintain and update DBDs, the performance deficiency is indicative of current plant performance. The finding is assigned a cross-cutting aspect in the area of Human Performance, in the component of Resources in that the licensee ensures that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety. Specifically, those necessary for complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components H.2(c).

Enforcement. 10 CFR Part 50 Appendix B, Criterion V, "Procedures," requires in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instruction, procedures, and drawings." Contrary to this, FCS has failed to maintain and update their DBDs in accordance with PED-QP-13. The finding has potential consequence in that FCS has a documented history of not ensuring all applicable regulatory requirements and the design basis are used in the production of quality documents. FCS is currently trying to evaluate, repair, maintain or modify systems, structures, components or procedures with processes that require accurate design information and historical perspective that was intended to be contained or referenced in the DBDs. Many FCS license required processes currently under NRC scrutiny; Design Control, Technical Specification compliance, 10 CFR 50.59, and 10 CFR 50.72 require accurate, assessable, and understandable Design Basis information. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into the licensee's corrective action program as 2012-17943: NCV 05000285/2012011-04, "Inadequate Design Basis Documentation."

4OA5 Other Activities

.1 (Discussed) NRC Temporary Instruction (TI) 2515/187, Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

a. Inspection Scope

Inspectors accompanied the licensee on a sampling basis, during their flooding walkdowns, to verify that the licensee's walkdown activities were conducted using the methodology endorsed by the NRC. These walkdowns are being performed at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC-endorsed walkdown methodology (ADAMS Accession No. ML12056A050). Nuclear Energy Industry (NEI) document 12-07 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC-endorsed methodology for assessing external flood protection and mitigation capabilities to verify that plant features, credited in the CLB for protection and mitigation from external flood events are available, functional, and properly maintained.

The inspectors accompanied the licensee on their walkdown of the Intake Structure and verified that the licensee confirmed the following flood protection features:

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

In addition to accompanying the licensee during their flooding walkdowns, the inspectors conducted their own independent walkdown to verify that the licensee adhered to their walkdown procedure. To select the area to walkdown independently and with the licensee, the inspectors considered areas that were determined to have a small available physical margin (APM). The inspectors also reviewed the scoping of the walkdown lists to ensure that all the components associated with a design basis flood were included in the licensee's walkdown list.

b. Findings

Failure to Properly Scope All the Pertinent External Flood Protection Features into the Walkdown List in Accordance with Industry Guidance NEI 12-07

Introduction: The inspectors identified a finding of very low safety significance (Green) for the licensee's failure to generate a complete inspection list, with all the external flood protection features credited in the current licensing basis documents for flooding events, to comply with NRC endorsed NEI 12-07, "Guidelines for Performing Walkdowns of Plant Flood Protection Features." Specifically, the scoping list did not include several active components, which are an essential part of Fort Calhoun's design basis flood mitigation strategy.

Description: The inspectors reviewed the licensee's inspection and walkdown documents associated with flooding reviews being performed in accordance with NEI-12-07 "Guidelines for Performing Walkdowns of Plant Flood Protection Features" at Fort Calhoun in response to a letter from the NRC to licensees, pursuant to 10 CFR 50.54(f). During the review, the inspectors identified that the licensee had completed their scoping of components for the temporary instruction, they had failed to properly scope all the flood protection features credited in the current licensing basis documents for flooding events. Specifically, while reviewing the Flooding Features Walkdown List, the inspectors identified that the six circulating water river sluice gates CW-14A through F, were not included. The licensee would later use that list to inspect and test design basis flood mitigating equipment in accordance with the NRC-endorsed guidance of NEI 12-07. The river sluice gates are an essential part of design basis flood mitigation strategy at Fort Calhoun Station. The sluice gates' function is to maintain the water level inside the Intake Structure by restricting the inflow to match the rate of pumped outflow in the raw water system. In the event of a design basis flood, the sluice gates are used to control the intake cell water level to prevent flooding of the raw water pump vaults. At Fort Calhoun Station, raw water pumps function as the safety related service water pumps that provide cooling to the component cooling water system and other safety related loads.

As a result of the inspectors' questions, the licensee's extent of condition review revealed that in addition to the sluice gates, 14 additional active components were improperly left out of the flooding features walkdown list. These included other active components that are accounted for in the licensee's design basis flood mitigating strategy such as the four raw water pumps, six drain valves in the radioactive waste disposal system and four drain isolation valves in the turbine building sump system. These drain valves are part of the design basis flood strategy in that they need to be closed to prevent flood propagation. The licensee's extent of condition review also identified that these drain valves were not part of a routine preventive maintenance (PM) program, since one did not exist for these kinds of valves. The licensee concluded that a preventive program would have to be developed since NEI 12-07 guidance states that components with an active function can be assumed to function properly if they are included in a routine PM or surveillance program and the testing performed under the program is acceptable.

Because the licensee did not follow the guidance in identifying critical active components that serve as flood barriers, these components were not scheduled for visual inspections or walkdowns. The licensee did not recognize they were needed to respond to the March 12, 2012, letter from the NRC to licensees requiring these reviews. The licensee acknowledged that they would not have identified these during subsequent reviews if the inspector did not identify it to them.

Analysis: The inspectors determined that a failure to properly scope all the flood protection features credited in the current licensing basis documents for flooding events as part of the NEI 12-07 flooding walkdowns was a performance deficiency. Using the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined this finding affected the Mitigating Systems cornerstone. The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of Protection Against External Factors (Flood Hazard) and it adversely affects the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, in addition to not scoping the sluice gates into the Flooding Features Walkdown List, fourteen additional active components such as the four raw water pumps and 10 drain valves would not have been scoped into the walkdown list. This would have prevented the licensee from identifying that preventive maintenance tasks needed to be created, and some active components that are an essential part of the flood mitigating strategy would not have been inspected and tested.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," and conducted a Phase 1 characterization and initial screening. Phase 1 initial screening determined that IMC 0609 Appendix A, Exhibit 2 "Mitigating Systems Screening Questions" should be used. Because the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding, or severe weather initiating event (e.g., seismic snubbers, flooding barriers, tornado doors), the finding screened as Green. This finding was related to the Work Practices component of the Human Performance cross cutting area because licensee personnel did not properly apply human error prevention techniques such as peer checking and proper documentation of activities (H.4(a)).

Enforcement: This finding does not involve enforcement action because no violations of NRC regulatory requirements were identified. Because this finding does not involve a violation and is of very low safety significance, it is identified as a FIN 05000285/2012011-05, "Failure to Properly Scope All the Pertinent External Flood Protection Features into the Walkdown List in Accordance with Industry Guidance NEI 12-07." The licensee entered the issue into the corrective action program (CAP) as CR

2012-14265, revised the scoping list accordingly and will implement a PM program for the appropriate valves.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On November 16, 2012, the inspectors presented the results of the radiation safety inspections to Mr. M. Prospero, Plant Manager, and other members of the licensee staff. 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 December 6, 2012, the inspectors presented the inspection results to Mr. L. Cortopassi, Site Vice President and Chief Nuclear Officer, 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

C. Cameron, Supervisor Regulatory Compliance
L. Cortopassi, Site Vice President
K. Erdman, Supervisor, Engineering Programs
M. Ferm, Manager, Site Performance Improvement
M. Frans, Manager, Engineering Programs
W. Hansher, Supervisor, Nuclear Licensing
K. Ihnen, Manager, Manager, Site Nuclear Oversight
J. James, Manager, Outage
R. King, Director, Site Maintenance
K. Kingston, Manager, Chemistry
T. Maine, Manager, Radiation Protection
E. Matzke, Senior Licensing Engineer
S. Miller, Manager, Design Engineering
V. Naschansy, Director, Site Engineering
T. Orth, Director, Site Work Management
A. Pallas, Manager, Shift Operations
M. Prospero, Division Manager, Plant Operations
T. Simpkin, Manager, Site Regulatory Assurance
M. Smith, Manager, Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285-2012-003-01	LER	Non-Conservative Error in Calculation for Alternate Hot Leg Injection Results in Unanalyzed Condition
05000285-2012-004-02	LER	Inadequate Analysis of Drift Affects Safety Related Equipment
05000285-2012-007-01	LER	Failure of Pressurizer Heater Sheath
05000285-2012-017-00	LER	Containment Valve Actuators Design Temperature Ratings Below those Required for Design Basis Accidents
05000285-2012-018-00	LER	Containment Air Cooling Units Operated Outside of Technical Specifications during Cycle 26
05000285-2012-019-00	LER	Traveling Screen Sluice Gates Found with Dual Indication

Opened and Closed

05000285-2012-011-01	NCV	Inadequate operability determination for containment internal structures

Opened and Closed

05000285-2012-011-02	NCV	Failure to Follow Radiation Work Permit Requirements
05000285-2012-011-03	NCV	Failure to Ensure that Adequate Equipment was Available to Measure River Level Locally to be Able to Comply with an Abnormal Operating Procedure
05000285-2012-011-04	NCV	Inadequate Design Basis Documentation
05000285-2012-011-05	FIN	Failure to Properly Scope All the Pertinent External Flood Protection Features into the Walkdown List in Accordance with Industry Guidance NEI 12-07

Closed

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
05000285-2012-007-00	LER	Failure of Pressurizer Heater Sheath

Discussed

TI 2515/187	TI	Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns

LIST OF DOCUMENTS REVIEWED

Section 1RO5: Fire Protection

PROCEDURES

NUMBER	TITLE	REVISION
SO-G-28	Standing Order, Station Fire Plan	83
SO-G-58	Standing Order, Control of Fire Protection System Impairments	37
SO-G-91	Standing Order, Control and Transportation of Combustible Materials	27
SO-G-102	Standing Order, Fire Protection Program Plan	12
SO-G-103	Standing Order, Fire Protection Operability Criteria and Surveillance Requirements	25

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
EA-FC-97-001	Fire hazards Analysis Manual	16

PROCEDURES

NUMBER	TITLE	REVISION
FC05814	UFHA Combustible Loading Calculation	11
USAR 9.11	Updated Safety Analysis Report, Fire Protection Systems	23

Section 1R15: Operability Evaluations

CONDITION REPORTS (CR)

2012-00550	2012-00551	2012-00657	2012-04392	2012-07085
2012-07413	2012-11933			

PROCEDURES

NUMBER	TITLE	REVISION
NOD-QP-31	Operability Determinations Process	51
QAP-5.1	Control of Plant Design and Modifications	12

DRAWINGS

NUMBER	TITLE	REVISION
11405-S-17	Reactor Plant Basement Floor Plan Elev 994Ft 0In, Outline	7
11405-S-23	Reactor Plant, Section and Detail, Outline	5

CALCULATIONS

NUMBER	TITLE	REVISION
EC 54436	Pipe Supports for CCW piping in Containment	0
EA 94-003	Alternate Seismic Criteria Methodologies	15

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
ACI-318	American Concrete Institute, ACI-318	1963 edition
USAR 5.11	Structures other than containment	10
USAR 5.5	Containment design criteria	6
USAR Appendix F	Classifications of structures and equipment and seismic criteria	9

Section 2RS01: Radiological Hazard Assessment and Exposure Controls

PROCEDURES

NUMBER	TITLE	REVISION
RPP	Radiation Protection Program	25
RP-202	Radiological Surveys	42
RP-204	Radiological Area Controls	62
RP-405	Radioactive Source Inventory Control	15

PROCEDURES

NUMBER	TITLE	REVISION
RP-ST-RM-0002	Surveillance Test Radioactive Material Sources Surveillance	8
SO-G-101	Standing Order; Radiation Worker Practices	38
RP-AD-200	Radiation Protection Surveillance Program	35
NMA-3	Nuclear Materials Accountability	18

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
11-QUA-052	Surveillance Report; Radiation Protection Operations	June 28, 2011
12-QUA-027	Assessment Report; Radiation Protection	April 26, 2012
RA #2010-1518	Radiation Protection Program; 2011 Self-Assessment Report	September 1, 2011

CONDITION REPORTS (CR)

2011-2521	2011-2624	2011-2644	2011-2702	2011-2742
2011-2820	2011-3034	2011-3036	2011-3156	2011-3179
2011-3237	2011-3239	2011-3290	2011-332	2011-3322
2011-3335	2011-3346	2011-3738	2011-3761	2011-3839
2011-4034	2011-4079	2011-4146	2011-4295	2011-4318
2011-4400	2011-4558	2011-4563	2011-4734	2011-4840
2011-6687	2011-6755	2011-6778	2011-6863	2011-7042
2011-8980	2011-9486	2012-00622	2012-02050	2012-02662
2012-02710	2012-02830	2012-03858	2012-05676	2012-06521
2012-07105	2012-09066	2012-09241	2012-09287	2012-10264
2012-12337	2012-13171	2012-13178	2012-13294	

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
	Dispositioned/Disposed Source Report	November 14, 2012
433967	Work Order Package; Inventory RHRA and VHRA Keys	July 12, 2012
44285-01	Work Order Package; Inventory RHRA and VHRA Keys	October 10, 2012
M-20120906-11	Radiological Survey; Room 5	September 6, 2012
M-20121018-2	Radiological Survey; Room 24, SFP Transfer Canal Pumps	October 18, 2012
M-20121101-2	Radiological Survey; Room 16	November 1, 2012

Section 2RS02: Occupational ALARA Planning and Controls

PROCEDURES

NUMBER	TITLE	REVISION
RP-AD-300	ALARA Program	28
RP-301	ALARA Planning/RWP Development and Control	46
RP-602	Personnel Dosimetry Issuance and Changeout	23
RP-670	Declared Pregnancy/Anticipated Pregnancy Procedure	0

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
11-QUA-040	Surveillance Report; ALARA Activities	May 19, 2011

CONDITION REPORTS

2011-2521	2011-2624	2011-2644	2011-2702	2011-2742
2011-2820	2011-3034	2011-3036	2011-3156	2011-3179
2011-3239	2011-3237	2011-3290	2011-3322	2011-332
2011-3335	2011-3346	2011-3738	2011-3761	2011-3839
2011-4034	2011-4079	2011-4146	2011-4295	2011-4318
2011-4400	2011-4558	2011-4563	2011-4734	2011-4840
2011-6755	2011-6778	2011-6863	2011-6687	2011-7042
2011-8980	2011-9486	2012-00622	2012-02050	2012-02662
2012-02710	2012-02830	2012-03858	2012-05676	2012-06521
2012-07105	2012-09066	2012-09241	2012-09287	2012-10264
2012-12337	2012-13171	2012-13178	2012-13294	

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE
	Five-Year Dose Reduction Plan	January 18, 2012
RWP 11-1507	ALARA Post-Job Review	June 1, 2012
RWP 11-2532	ALARA Post-Job Review	April 12, 2012
RWP 11-1522	ALARA Post-Job Review	September 19, 2011
	ALARA Committee Meeting Minutes	December 15, 2011
	ALARA Committee Meeting Minutes	March 15, 2012
	ALARA Committee Meeting Minutes	June 28, 2012

Section 4OA1: Performance Indicator Verification

CONDITION REPORTS (CR)

2010-2387	2011-2162	2011-3651	2011-5414	2011-7496
2011-10129	2012-01021	2012-01324	2012-02430	2012-03718
2012-04392	2012-04594	2012-04825	2012-07085	2012-08621

PROCEDURES

NUMBER	TITLE	REVISION / DATE
RP-901	Evaluating Program Effectiveness	8
RP-907	Radiological Analysis	3

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Various Operator Logs	10/1/2011 to 9/30/2012
NEI 99-02	Regulatory Assessment Performance Indicator Guideline	6

Section 4OA3: Event Follow-Up

CONDITION REPORTS (CR)

2012-01914	2012-02430	2012-04327	2012-10206

LICENSEE EVENT REPORTS (LER)

NUMBER	TITLE	REVISION / DATE
2012-003	Non-Conservative Error in Calculation for Alternate Hot Leg Injection Results in Unanalyzed Condition	1
2012-004	Inadequate Analysis of Drift Affects Safety Related Equipment	2
2012-007	Failure of Pressurizer Heater Sheath	1
2012-019	Traveling Screen Sluice Gates Found with Dual Indication	0

Section 4OA4: IMC 0350 Inspection Activities

CONDITION REPORTS (CR)

2011-6251	2011-6812	2011-6750	2011-6557	2011-5463
2011-6920	2011-8682	2011-7366	2011-6598	2011-6622
2011-7696	2012-11168	2012-13312	2012-10206	2012-17078
2012-04425	2012-16232	2012-16428	2012-09996	2012-15293
2012-09193	2012-15216	2012-01655	2012-18359	2012-07729
2011-6757	2012-18388	2012-05949	2012-07722	2012-18369
2012-18390	2012-18392	2012-18355	2012-05967	2012-01947
2012-18354	2012-18401	2012-18357	2012-10477	2012-18361

CONDITION REPORTS (CR)

2012-18362	2012-01655	2012-18359	2012-07729	2012-04070
2012-15811	2011-6621	2012-05854	2012-05850	2012-17473
2012-14638	2011-5414	2012-06508	2012-05854	2012-03921
2012-12451	2012-04460	2012-13815	2012-00600	2012-12270
2012-00131	2012-12278	2012-06531	2012-13694	2012-01820
2012-00552	2012-04091	2012-12273	2012-15278	2012-13552
2012-01805	2012-01114	2012-12280	2012-10510	2012-04050
2012-03816	2012-04461	2012-14793	2012-14082	2012-12279
2012-10448	2008-3180	199600902	2011-2042	2012-08125

WORK ORDERS (WO)

408067	399136	400207	402965	415069
403024	442408	429857	427647	399047
415118	427174	427175	419609	420421
405617	421709	426512	434547	455695
441791	440162			

WORK REQUESTS (WR)

170469	170468	170470	170474

PROCEDURES

NUMBER	TITLE	REVISION
OP-PM-FW-0004	Third Auxiliary Feedwater Pump Operability Verification	36
PE-OM-FO-1000	Portable Filtering of Fuel Oil Storage Tanks	5
PLDBD-CS-56	External Flooding	1
USAR 9.8	Auxiliary Systems: Raw Water System	31
PE-RR-AE-1000	Flood Barrier Inspection and Repair	9
PE-RR-AE-1001	Flood Barrier and Sandbag Staging and Installation	16
PE-RR-AE-1002	Installation of Portable Steam Generator Make-up Pumps	5

CONDITION REPORTS (CR)

FCSG-64	External Flooding of Site	2
SO-G-124	Flood Barrier Impairment	2
AOP-01	Acts of Nature	31
AOP 38	Blair Water Main Trouble	4
AOP-36	Loss of Spent Fuel Cooling	8
AOP-19	Loss of Shutdown Cooling	17
AOP-18	Loss of Raw Water	7
OI-CW-1	Circulating Water System Normal Operation	67
PED-SEI-46	Functional Equipment Group and Functional Importance Determination Process	2
SO-M-2	Preventive Maintenance Program	45
PED-SEI-13	Preventive Maintenance Program – Technical Basis	14
MRII-3.1	Maintenance Rule Implementing Instruction	2
SO-G-23	Surveillance Test Program	59
SO-R-1	Reportability Determinations	27
SDBD-AC-RW-101	Raw Water	39
PED-SEI-34	Maintenance Rule Program	9
FCSG-20	Abnormal Operating Procedure and Emergency Operating Procedure Writer's Guide	10
SDBD-STRUC-503	Intake Structure	12
NOD-QP-3	10 CFR 50.59 and 10 CFR 72.48 Reviews	32a
OI-RW-2	Raw Water System Outage for Maintenance	18
OI-RW-1	Raw Water Normal Operation	104
SAP-29	Severe Weather and Flooding	13
PBD-19	Electrical Equipment Qualification Program	4
PED-QP-15	Electrical Equipment Qualification Program	12
NP-95003-KAR-AD	EC 45086, 4160V Breaker Replacement	3
NP-95003-KAR-AD	EC 33464, Replace AK-50 Main and Bus-Tie Breakers	3
ERPG-CQE-01	Engineering Recovery Process Guide CQE Part Replacement Review	0

CONDITION REPORTS (CR)

NOD-QP-1	Preparation, Approval and Distribution of NOD Documents	39a
PED-QP-9	Setpoint Control	5
NOD-QP-28	Safety Enhancement Program	6
QAP-3.4	Records Management	12
PED-QP-11	Independent Design Verification (IDV) and Independent Review of Configuration Changes	10
PED-QP-10	Document Control and Configuration Management	10a
PED-QP-5	Engineering Analysis Preparation, Review and Approval	41
PED-QP-3	Calculation Preparation, Review and Approval	33
PED-QP-2	Configuration Change Control	56
PED-QP-13	Design Basis Document Control	7

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
EC53240	Install/Remove Fuel Oil Pipe Extension on FO-1	6/8/11
FC08040	Diesel Fuel Oil Tanks FO-1 and FO-10 Buoyancy Calculation for Structural Integrity	0
EC53939	Modification of HE-6A/B to Withstand Design Basis Flood	0
SAO-12-001	Operability of Sluice Gates	
TD B580.0430	Installation, Operation and Maintenance Instructions for Byron Jackson 2 Stage Raw Water Pumps	
FC 08030	Intake Structure Cell Level Control Using the Intake Structure Sluice Gates	11
EA-FC-06-032	Environmental Parameters for Electrical Equipment Qualification	0
EA-FC-10-020	Electrical Equipment Qualification Radiation Dose Reconstitution Analysis	0
EA-11-037	Summary of Design Basis Reconstitution for High Energy Line Break (HELB) Outside of Containment in Response to CR 2007-3407	0
EA-FC-12-005	Harsh-Mild Environment Threshold Criteria	0

CONDITION REPORTS (CR)

USAR Appendix G	Responses to 70 Criteria	22
CERN 89-12	Compilation Of Radiation Damage Test Data, Part I, 2nd Edition: Halogen-Free Cable-Insulating Materials	1989
IPS-701	Thermal Capability Curves (I^2t) For Conax Electric Penetration Assemblies and Electric Conductor Seal Assemblies	7/16/1981
EGS-TR-23047-81	Test Procedure for Pressure Boundary Capability of As-Installed Feedthroughs for Electrical Penetration Assemblies at Fort Calhoun Station	A
ORNL-TM-1757	The Effect of Air on the Radiation-Induced Degradation of Polytetraflouroethylene (Teflon)	1967
LIC-84-0121 (WIP44564)	Environmental Qualification of Safety Related Electrical Equipment	5/31/1984
	Safety Evaluation for Final Resolution of Environmental Qualification of Electric Equipment Important to Safety – Fort Calhoun Unit 1	02/15/1985
LIC-85-009	Environmental Qualification of Safety-Related Electrical Equipment	01/10/1985
USAR-Appendix M	Postulated High Energy Line Repture Outside the Containment	10
NLI Report RCA-09315397-1	OPPD Switchgear Arc Flash Root Cause Analysis	Revision 0/ August 2012
50-285/85-22	Safety Systems Outage Modification Inspection (Design)	1/21/1986
50-285/85-29	Safety Systems Outage Modification Inspection (Installation and Test)	3/19/1986
LIC-86-106	50-285/85-22 Response	4/15/1986
LIC-86-192	50-285/85-29 Response	5/22/1986
50-285/88-200	Re-inspection of Safety Systems Outage Modification Inspection Design Findings	9/16/1988
LIC-87-691	Update of Response to Notice of violation concerning Safety System Outage Modification Inspection (SSOMI)	12/23/1987
EA 86-176	Notice of Violation and Proposed Imposition of Civil Penalty	1/26/1987
	NRC to OPPD enforcement conference letter follow up	2/12/1987
LIC-87-086		4/10/1987

CONDITION REPORTS (CR)

LIC-88-1029	Response to 50-285/88-200	12/31/1988
LIC-88-1094	Safety Enhancement Program	12/9/1988
50-285/88-201	Fort Calhoun Operational Safety Team Inspection	2/9/1989
	NRC Safety Enhancement Program Assessment	6/20/1989
	NRC Safety Enhancement Program Assessment	10/31/1989
LIC-89-1006	Safety Enhancement Program One Time Commitments	11/6/1989
	NRC Safety Enhancement Program Assessment	5/24/1990
50-285/91-13	NRC Inspection Report No. 50-285/91-13	5/15/1991
NPM 2.02	Nuclear Policy Manual - Safety Enhancement Program	2

Section 4OA5: Other Activities

CONDITION REPORTS (CR)

2012-14265	2012-15194	2012-16864	2012-16884	2012-16891
2012-16901				

PROCEDURES

NUMBER	TITLE	REVISION
PLDBD-CS-56	External Flooding	1
USAR 9.8	Auxiliary Systems: Raw Water System	31
PE-RR-AE-1000	Flood Barrier Inspection and Repair	9
PE-RR-AE-1001	Flood Barrier and Sandbag Staging and Installation	16
PE-RR-AE-1002	Installation of Portable Steam Generator Make-up Pumps	5
FCSG-64	External Flooding of Site	2
AOP-01	Acts of Nature	31
SDBD-STRUC-503	Intake Structure	12

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Flood Protection Features Walkdown List	10/04/2012
NEI 12-07	Walkdown Record Forms for West Wall of Intake Structure	09/21/2012
OPPD Training	Walkdown and Procedure Review Indoctrination	08/2012
NEI 12-07	Walkdown Record Forms for East Wall of Air Compressor Room	09/21/2012