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

November 13, 2012

EA-12-174

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

Subject: FORT CALHOUN - NRC INTEGRATED INSPECTION REPORT NUMBER 05000285/2012005, AND NOTICE OF VIOLATION

Dear Mr. Cortopassi:

On September 30, 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 October 18, 2012, with Mike Prospero, Plant Manager, and other members of your staff, and on November 7, 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.

Based on the results of this inspection a Severity Level IV violation of NRC requirements was identified involving the failure to update the Updated Safety Analyis Report. This violation was evaluated in accordance with the NRC Enforcement Policy. The violation is being cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in detail in the subject inspection report. The violation is being treated as a cited violation, consistent with Section 2.3.2(a)(3) of the NRC Enforcement Policy. Specifically, this violation was repetitive as a result of ineffective corrective actions and was identified by the NRC.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements."

One NRC identified finding of very low safety significance (Green) was identified during this inspection. This finding was determined to involve a violation of NRC requirements. The NRC is treating this violation as a noncited violation consistent with Section 2.3.2 of the Enforcement Policy.

If you contest these 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/

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

Docket: 50-285
License: DPR-40

Enclosures:
1. Notice of Violation
2. NRC Inspection Report 05000285/2012005
   w/Attachment: Supplemental Information

cc w/ encl: Electronic Distribution

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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	SPE:DRP/F	SPE:DRP/F	C:DRS/PSB2	C:ORA/ACES	BC:DRP/.F
JCKirkland	JFWingebach	RWDeese	JDrake	HGepford	MCHay
/RA via E/	***/RA via E/***	***/RA/***	***/RA/***	***/CYoung for/***	***/RA/***
11/13/12	11/13/12	11/8/12	11/13/12	11/13/12	11/13/12

NOTICE OF VIOLATION

Omaha Public Power District Fort Calhoun Station	Docket No.: 05000285 License No.: DPR-40 EA-12-174

During an NRC inspection conducted from June 18 to August 3, 2012, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10 CFR 50.71(e) requires, in part, that each person licensed to operate a nuclear power reactor under the provisions of 50.21 or 50.22, shall update periodically the final safety analysis report (FSAR) originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. The submittal shall include the effects of all changes made in the facility or procedures as described in the FSAR; and all safety analyses and evaluations performed by the applicant or licensee either in support of approved license amendments or in support of conclusions that changes did not require a license amendment in accordance with § 50.59(c)(2). The updated information shall be appropriately located within the update to the FSAR.
>
> Contrary to the above, from December 2006 to June 2012, the licensee failed to assure that the information included in the Updated Safety Analysis Report contains the latest information developed, including the effects of all changes made in the facility or procedures as described in the Report. Specifically, since December 2006, the licensee stored a significant source of radioactivity in the Original Steam Generator Storage Facility but failed to describe the volume of waste, the principal sources of radioactivity, the total quantity of radioactivity, and the estimated dose rate at the site boundary per curie of radioactivity in the Updated Safety Analysis Report.

This is a Severity Level IV violation (Section 6.1.d).

Pursuant to the provisions of 10 CFR 2.201, Omaha Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector - Fort Calhoun Station, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-12-0174" and should include for each violation: (1) the reason for the violation or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days of receipt.

Dated this 13th day of November 2012

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2012005
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	August 19 through September 30, 2012
Inspectors:	J. Kirkland, Senior Resident Inspector J. Wingebach, Resident Inspector A. Klett, Reactor Operations Engineer A. Rosebrook, Senior Project Engineer R. Deese, Senior Project Engineer F. Ramirez, Resident Inspector K. Clayton, Senior Operations Engineer W. Smith, Project Engineer A. Fairbanks, Reactor Inspector
Approved By:	Michael Hay, Chief, Project Branch F Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2012005; 08/19/2012 – 09/30/2012; Fort Calhoun Station, Integrated Resident, Inservice Inspection, and Confirmatory Action Letter Report

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

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The NRC identified a noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to take timely corrective actions with respect to nonconforming conditions in several circuit breakers. These conditions were determined to have been the cause of the 1B4A bus bar failure that initiated a fire on June 7, 2011. These conditions were not corrected in a timely manner and the licensee continued to operate with a degraded breaker for nine months after the breaker tripped unexpectedly during the June 7, 2011, fire event. The licensee entered this issue into their corrective action program as CRs 2012-01884 and 2011-5414.

  The violation was determined to be more than minor because it affected the Initiating Events Cornerstone attribute of protection against external events (i.e., fire). The issue adversely affected the associated cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations because the condition that contributed to the fire event was left uncorrected. The finding screened to Green in accordance with IMC 0609, Appendix G because RCS makeup capability was not degraded. The inspectors determined that the issue had a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program (P.1(d)). (4OA4.1.c.(3).2).

Cornerstone: Miscellaneous

- SLIV. The inspectors identified a cited violation of 10 CFR 50.71(e), "Maintenance of Records, Making of Reports," for the failure to update the Updated Safety Analysis Report with a detailed description of the Original Steam Generator Storage Facility. Specifically, since December 2006, the licensee stored a significant source of radioactivity in the Original Steam Generator Storage Facility, but failed to describe the volume of waste, the principal sources of radioactivity, the total quantity of radioactivity, and the estimated dose rate at the site boundary per curie of radioactivity in the Updated Safety Analysis Report. The licensee has entered this violation into their corrective action program as Condition Report 2012-05725.

This issue was evaluated using traditional enforcement because it has the potential to impact the NRC's ability to perform its regulatory function. This issue is being characterized as a Severity Level IV violation in accordance with Section 6.1.d.3 of the NRC Enforcement Policy. Cross-cutting aspects are not assigned to traditional enforcement violations (Section 2RS08).

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

The station remained in Mode 5 with the fuel in the reactor vessel for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

During August through November, 2012, the inspectors completed a focused inspection of the steam generators in response to the San Onofre Nuclear Generating Station (SONGS) primary to secondary steam generator leakage. The inspection focused on similarities of steam generator design, and verified that the types of degradation affecting SONGS steam generators does not impact the steam generators at the Fort Calhoun Station. The inspection focused on:

- Retainer and freespan indications.
- Adequacy of Mitsubishi's thermal-hydraulic model.
- Refueling outage eddy current testing results.
- 10 CFR 50.59 review

The inspectors reviewed the updated safety analysis report (USAR), steam generator design documents, eddy current testing (ECT) procedures and data results, corrective actions, and performed a walkdown of the steam generators. The inspectors also attended a presentation provided to the licensee by Mitsubishi Heavy Industries (MHI). Specifically, the inspectors reviewed:

- 10 CFR 50.59 evaluation of the replacement steam generators.
- Eddy current examination reports for the 2008 refueling outage.
- Secondary inspection results for the 2008 refueling outage.
- MHI presentation that included discussions on retainer bar random vibrations, and in-plane flow elastic instabilities of tube-to-tube wear.
- Fort Calhoun steam generator long term inspection strategy plan.
- Westinghouse second review of eddy current testing data.
- Independent review of raw ECT data on EddyNet format

Industry experience has shown that most deficiencies in steam generator design are typically identified during eddy current inspections following the first operating cycle. The steam generators at Fort Calhoun Station were replaced in 2006, and inspected during the refueling outage in 2008. NRC experts performed an independent review of select ECT raw data, with an emphasis on low frequencies absolute data channels indicative of tube-to-tube wear, with no issues identified. As a result of the information presented to the NRC, including a Westinghouse second review of ECT data, the NRC's independent review of the same data, and positive industry experience of steam generator designs not experiencing issues after a successful first cycle inspection, the inspectors determined that reasonable assurance exists that the degradation mechanism experienced in SONGS steam generators does not exist at this time for the Fort Calhoun Station.

The inspectors will review the following documentation as it becomes available:

- Revised degradation and operational assessment.

Certain aspects documented as open items in the SONGS Augmented Inspection Team report (ML12188A748) have the potential to require further inspections at Fort Calhoun Station.

b. Findings

No findings were identified.

1R15 Operability Evaluations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following assessments:

- August 31, 2012, Operability of the reactor cavity walls prior to moving fuel from the reactor vessel to the spent fuel pool

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 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 as defined in Inspection Procedure 71111.15-05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

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

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

- August 28, 2012, OP-ST-FH-0005, Refueling System Spent Fuel Handling Machine Refueling Interlocks Test

- September 1, 2012, OP-ST-FH-0002, Refueling System Fuel Transfer System Interlocks Test
- September 4, 2012, OP-ST-FH-0001, Refueling System Fuel Handling Machine (FH-1) Interlocks Test

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

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

b. Findings

No findings were identified.

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

a. Inspection Scope

This area was inspected to verify the effectiveness of the licensees programs for updates to the Updated Safety Analyiss Report related to the processing, handling, and storage of radioactive material.

b. Findings

(1) Failure to Update the Updated Safety Analysis Report-Solid Wastes

Introduction. The inspectors identified a Severity Level IV violation of 10 CFR 50.71(e), “Maintenance of Records, Making of Reports”, for failure to update the Updated Safety Analysis Report with information about the Original Steam Generator Storage Facility that was constructed in 2006 for long-term storage of large decommissioned components .

Description. In 2006, the licensee built the Original Steam Generator Storage Facility for long-term solid radioactive waste storage of the two original steam generators, the pressurizer, the reactor vessel head, and four concrete reactor vessel head missile shield blocks. From the licensee’s estimation, the Original Steam Generator Storage Facility contained 404 curies. However, this significant source of radioactivity was not described in the licensee’s Updated Safety Analysis Report. On November 10, 2010, the NRC identified a Severity Level IV noncited violation for the failure to update the Updated Safety Analysis Report per 10 CFR 50.71(e) because the licensee had not described the Original Steam Generator Storage Facility in the Updated Safety Analysis Report (NCV 05000285/2010004-03).

During the June 2012 radiation protection inspection, the inspectors toured the Original Steam Generator Storage Facility and reviewed the licensee’s implementation of corrective actions associated with the previous violation. The licensee’s corrective actions for the 2010 noncited violation were initially addressed in Condition Report 2010-03636 and included an apparent cause analysis. The licensee’s apparent cause for the violation stated that the “Engineering Change Package was developed to an unknowable or changing [NRC] requirement.” The condition report further stated that “this violation showed a common misapplication of the regulations related to storage which may have been in place for several years.” The condition report also stated that engineering will be contacted to perform a 10 CFR 50.59 screening and update the USAR by January 2011. However, the inspectors determined that the licensee did not implement corrective actions based on the noncited violation as addressed in Condition Report 2010-03636. In 2011, the licensee did not update the Updated Safety Analysis Report to describe the Original Steam Generator Storage Facility.

Prior to the June 2012 inspection, the licensee performed a self-assessment as part of Condition Report 2012-03704 to verify that Chapter 11 of the Updated Safety Analysis Report had been updated, including a description of the Original Steam Generator Storage Facility. Based on the self-assessment results, the licensee submitted a revision to the Updated Safety Analysis Report Chapter 11.2.4.1, “Radioactive Waste Storage” to the NRC in June 2012. The inspectors’ review determined that the information added in the June 2012 revision of the Updated Safety Analysis Report was inadequate. The licensee’s update in Chapter 11.2.4.1, of the Updated Safety Analysis Report merely stated that radwaste waiting disposal is stored in the Original Steam Generator Storage Facility located on the west side of the plant site, north of the main access road. The inspectors concluded that the Original Steam Generator Storage Facility was being used to store a significant source of radioactivity that was not adequately described in Chapter 11 of the licensee’s Updated Safety Analysis Report. Some of the information missing about the Original Steam Generator Storage Facility included the volume of waste, the principal sources of radioactivity, the total quantity of stored radioactivity, and the estimated dose rate at the site boundary per curie of stored waste.

As of June 22, 2012, the inspectors concluded that the corrective actions implemented in Condition Report 2010-03636 for the 2010 violation and the self-assessment under Condition Report 2012-03704 were inadequate to comply with 10 CFR 50.71(e), in that, Chapter 11 of the Updated Final Safety Analysis Report did not adequately describe the Original Steam Generator Storage Facility. This issue was entered into the licensee’s corrective action program as Condition Report 2012-05725.

<u>Analysis</u>. Failure to update the Updated Safety Analysis Report as required by 10 CFR 50.71(e) with a detailed description of the Original Steam Generator Storage Facility was a performance deficiency. This issue was evaluated using traditional enforcement because it had the potential to impact the NRC’s ability to perform its regulatory function. The issue was characterized as a Severity Level IV violation in accordance with Section 6.1.d.3 of the NRC Enforcement Policy, in that, the erroneous [incomplete] information in the Final Safety Analysis Report Update was not used to make an unacceptable change to the facility or procedures. Cross-cutting aspects are not assigned to traditional enforcement violations.

Enforcement. 10 CFR 50.71(e), "Maintenance of Records, Making of Reports," states, in part, that each person licensed to operate a nuclear power reactor shall update periodically the Updated Safety Analysis Report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. Contrary to the above, from December 2006 to June 2012, the licensee failed to assure that the information included in the Updated Safety Analysis Report contains the latest information developed to include the effects of all changes made in the facility. Specifically, since December 2006, the licensee stored a significant source of radioactivity in the Original Steam Generator Storage Facility, but failed to describe the volume of waste, the principal sources of radioactivity, the total quantity of radioactivity, and the estimated dose rate at the site boundary per curie of radioactivity in the Updated Safety Analysis Report. This violation is being treated as a cited violation, consistent with Section 2.3.2(a)(3) of the NRC Enforcement Policy: NOV 05000285/2012005-01, "Failure to Update the Updated Safety Analysis Report-Solid Waste."

4. OTHER ACTIVITIES

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

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

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

.1 (Open) Licensee Event Report 05000285/2011-010-00: Fire Causes a Circuit Breaker to Open Outside Design Assumptions

On June 7, 2011, a bus fault in load center 1B4A initiated a switch gear fire that resulted in the opening of a circuit breaker which supplies power to load center 1B3A, associated with the opposite train. A fire in one fire area that resulted in a loss of power to a load center associated with the opposite train is not in compliance with 10 CFR 50, Appendix R. The analysis assumes that a fire in a fire area affecting one train of power will be isolated such that power associated with the redundant train will be maintained.

A root cause analysis is being performed to determine the cause of the failure.

The affected bus was de-energized and the Halon system extinguished the fire. The Halon system was recharged and restored to service. Inspections and testing of the unaffected 480 V buses, the supply circuit breakers to the 480 V buses, and the 480 V bus tie circuit breakers were performed. Appropriate 480 V supply circuit breakers and bus tie circuit breakers passed their inspections and testing. The fire damaged switchgear (1B4A), which contains two 480V supply circuit breakers, 1B4A and BT-1B4A (supply circuit breaker to the associated "island" bus), is being replaced. Additional corrective actions will be specified following the completion of the root cause analysis.

.2 (Open) Licensee Event Report 05000285/2012-014-00: Containment Beam 22 Loading Conditions Outside of the Allowable Limits

On July 11, 2012, while performing the Extent of Condition for an existing Condition Report (CR) it was determined that Beam B-22, a structural member of the containment internal structure at the 1013 foot elevation, loading conditions were outside the allowable limits for both Working Stress and No Loss of Function load combinations as noted in the USAR Section 5.11. This condition was identified on July 11, 2011, while the unit was shutdown and reported to the U.S. Nuclear Regulatory Commission (NRC) Headquarters Operations Center the same day at approximately 1603 CDT under Event Notification Number 48094.

A cause analysis is being evaluated and will be published in a supplement to this LER.

.3 (Open) Licensee Event Report 05000285/2012-015-00: Electrical Equipment Impacted by High Energy Line Break Outside of Containment

While reviewing a draft of the Master List Reconstitution for Electrical Equipment Qualification (EA-FC-08-011), Fort Calhoun Station (FCS) Engineering Department identified that some of the listed components may not be qualified for the environments where they are located. This was discovered during a comprehensive re-evaluation of potential high energy line breaks and radiological impacts outside containment initiated in response to issues identified by the station staff. This condition was identified on September 16, 2011, while the unit was shutdown.

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

.4 (Open) Licensee Event Report 05000285/2012-016-00: Unanalyzed Charging System Socket Welds to the Reactor Coolant System

On July 17, 2012, Fort Calhoun Station (FCS) identified a deficiency as part of the analyses being performed in support of resolution to the question as to whether some Class I pipe was potentially not qualified as Class 1. Condition Report (CR) 2012-07724 documented that preliminary results from an Thermal Fatigue Analysis on the chemical and volume control system (CVCS) concluded that; 1) The 2 inch socket welded fittings on Reactor Coolant System (RCS) branch line piping cannot be qualified, and 2) The 2 inch charging lines are considered to be in an unanalyzed condition exceeding thermal cycle fatigue and seriously degraded.

A cause analysis was completed and determined that the CVCS Class 1 piping was constructed using socket welded fittings.

CVCS was declared inoperable. The normal charging headers to the RCS are classified as inoperable until further evaluations or required repairs are performed. CVCS has been isolated to prevent any further thermal transients to the suspect welds. In addition, the affected waste disposal piping line which was scoped under the extent of condition is being addressed under CR 2012-12184. Contingency actions have already been taken to secure the letdown line so no thermal stress may be introduced to those socket welds. The affected welds will be replaced prior to plant heatup.

4OA4 IMC 0350 Inspection Activities (92702)

Inspectors began the 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 the beginning phase of this inspection is to assess the licensee's performance and progress in addressing its implementation and effectiveness of FCS's Integrated Performance Improvement Plan (IPIP), significant performance issues, weaknesses in programs and processes, and flood restoration activities. This phase of inspection determines whether the depth and breadth of performance concerns are understood.

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

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

.1 Causes of Significant Performance Deficiencies and Assessment of Organizational Effectiveness

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 Fort Calhoun Station to maintain procedures and equipment that protects the plant from the effects of a design basis flood. These deficiencies resulted in a Yellow (substantial safety significance) finding.

(1) 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.

The onsite activities included specific walk-downs of licensee procedure to mitigate flooding such as PE-RR-AE-1001, "Flood Barrier and Sandbag Staging and Installation", PE-RR-AE-1002, "Installation of Portable Steam Generator Pumps," Abnormal Operating Procedure (AOP)-1, "Acts of Nature" Section I, "Flood," and OI-CW-1, "Circulating Water System Normal Operation," Attachment 18, "Sand IntrusionMitigation." In addition, the inspectors completed a more detailed walk-down of the intake structure and pre-staged flooding equipment; interviews with personnel involved in the flooding emergency preparedness and recovery efforts; and observation of recovery effort meetings. The in-office activities consisted of reviews of documents associated with the recovery efforts, procedures associated with flooding mitigation strategies, system lesson plans, and condition reports.

(2) Assessment

The inspectors' review focused mainly on the adequacy of procedures that are associated with mitigation strategies for a design basis flood. As a result of the various procedure walk-downs, the inspectors had observations associated with procedure sustainability and quality. For example, PE-RR-AE-1001, Attachment 23, "Fuel Transfer Hose to Emergency Diesel Generator (EDG) Day Tanks," does not prescribe a specific plan to route the EDG fuel transfer hose. This procedure attachment is used to provide the EDG day tanks with fuel in the event elevated river levels were expected to last longer than 7 days. The inspectors identified that the procedure did not contain detailed information regarding how the hose would be routed from the tanker at the entrance of the plant to the EDG day tanks to ensure it will not be damaged by other plant traffic. This observation was provided to the licensee and was placed in the Corrective Action Program.

During the walk-down of the flooding procedures listed in the scope of this report section, the inspectors also noted that, even though the main pieces of equipment and tools listed in the procedures were pre-staged, some of the smaller tools were not. The inspectors noted that if the licensee had a more meticulous pre-staging of equipment including small tools and consumables, the number of trips to the tool room would be minimized and flood preparations would be more efficient. The licensee entered this observation into the corrective action process.

During this inspection period, the inspectors assessed the flood preparations associated with the emergency response facilities such as the Technical Support Center (TSC), the Operations Support Center (OSC), and the Emergency Operations Facility (EOF). The inspectors reviewed the licensee's plan to provide for an alternate emergency response facility in case the original locations were expected to flood.

The licensee was able to demonstrate that an adequate plan existed for alternate TSC, OSC, and EOF facilities in case of a flood. In addition, the inspectors noted that there are no thresholds to transfer the TCS and OSC to alternate locations. The inspectors also noted a general lack of rigor and details in the procedures to respond to prepare and respond to a flood. Specifically, the inspectors noted the licensee did not have detailed plan on managing the distribution of resources and personnel, and the strategy during the preparation time for an imminent flood. As a result of the inspectors' observations, the licensee is currently constructing a resource-loaded schedule that delineates the different tasks and times requested for all the preparations needed prior to a flood. The inspectors will review the plan and continue to have further discussions with licensee operations and emergnency preparedness personnel. Further in-depth Emergency Preparedness (EP) inspections will be performed by EP inspectors and will be documented in the future as part of Restart Checklist Item 5.f.

The inspectors reviewed the basis for the number of hours that the licensee bases their entire flooding planning on. The inspectors wanted to ensure that the technical foundation for that window of preparation time was adequate and that the licensee would still be able to stage equipment, stack sand bags, and assemble flood barriers in enough time before the plant grounds start to flood.

(3) Findings

No findings of significance were identified.

.b Reactor Protection System contact Failure – White Finding

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

(1) Inspection Scope

The NRC inspected and will continue to inspect the root cause, extent of cause, and extent of condition related to the contactor failure and the associated process failures. The on-site activities included interviews and discussions with staff performing evaluations of significant performance issues, programs, and processes; and observation of conduct of recovery effort meetings.

(2) Assessment

The team completed the review of revision 2 of the Root Cause Analysis for the contactor failure, RCA 2011-0451, during previous inspection weeks. However, no progress was made in this area during the six weeks of this reporting period because the licensee started a new root cause analysis (revision 3) the week of September 24, 2012. Revision 3 of this root cause analysis will supersede the previous two versions because of errors, omissions, and poor clarity. The licensee continues to try to pull this date up for completion of the root cause itself to November 9, 2012, and does not currently have a schedule for completion of all corrective actions. The NRC will close out this issue for restart after the inspections verify that the station has 1) completed revision 3 of this root cause analysis, 2) completed the corrective actions from the root cause analysis, and 3) completed all actions necessary to prevent re-occurrence.

(3) Findings

No Findings of significance were identified.

.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 caused a fire in the safety-related 480 volt (V) electrical switchgear. These deficiencies resulted in a finding having Red (i.e., high) safety significance.

(1) Inspection Scope

During the inspection period, the NRC assessed (and will continue to assess during upcoming inspection periods) the licensee's root cause, extent of cause, and extent of condition evaluations related to the fire and associated equipment and process failures.

The on-site activities included a walk-down of the remains of the breaker that was on fire and a tour of the switchgear rooms, interviews and discussions with licensee staff, and observation of recovery effort meetings. 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.

The team also reviewed modification EC 33464, "Replace AK-50 480 V Main and Bus-Tie Breakers With Molded Case Type or Equivalent," Revision 0, which replaced 12 General Electric AK-50 low voltage power circuit breakers with Nuclear Logistics Incorporated/Square-D Masterpact circuit breakers / cradle assemblies, and digital trip devices in November 2009. The modification replaced six feeder circuit breakers and six bus-tie breakers.

The team interviewed the system engineers responsible for the 480 VAC distribution system and electrical maintenance technicians that maintained the system. The team interviewed operations personnel and discussed procedures and training for the modification. The team reviewed the modification to determine if the requirements of 10 CFR 50.59, "Changes, Tests and Experiments," were met, including understanding the possible failure modes, and to assess the post-modification testing completeness for cradle and breaker positioning, electrical resistance, and other critical parameters.

(2) Assessment

As previously discussed in NRC IR 050-00285/2012004, when evaluating whether a risk significant finding may be closed, NRC IP 95002 directs inspectors to :

1) To provide assurance that the root and contributing causes of individual and collective (multiple greater than green inputs) risk-significant performance issues are understood.
2) To independently assess and provide assurance that the extent of condition and the extent of cause of individual and collective (multiple greater than green inputs) risk-significant performance issues are identified.
3) To independently determine if safety culture components caused or significantly contributed to the individual and collective (multiple greater than green inputs) risk-significant performance issues.
4) To provide assurance that a licensee's corrective actions for risk-significant performance issues are sufficient to address the root and contributing causes and prevent recurrence.

In order to achieve these objectives the inspectors independently reviewed the licensee's evaluations of the event and determined that the following conditions contributed to 1) the initiation of the fire event or 2) the unexpected system response to the initiating event. In the inspectors' assessment are SCAQs based upon the OPPD QA manual's definition since each of these conditions would have precluded plant response to the event from ending up outside plant design basis and resulting in a high safety significance (RED) finding.

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

1) The postulated cause of the fire was a high impedance connection between the breaker cradle assembly and the 480 VAC bus stabs which caused localized overheating and the bus bar failure, which initiated the event. This

condition was the focus of CR 2011-5414. The licensee's corrective actions developed included replacing the damaged switchgear components, correcting and/or verifying the alignment of the remaining breaker and cradle assemblies, correcting the silver plating on all the breaker stabs, and revising design procedures. The inspectors reviewed the corrective actions completed and planned and concluded they were adequate to preclude repetition of this SCAQ. However, NLI and Square D (the vendors for the breaker) completed an independent RCA for the event on 8/22/12. As of the end of the inspection period the NRC had not completed its review of this independent RCA.

2) During the fire, a phase-to-phase arc fault occurred for 42 seconds, which generated a fault current value of 16,000 amperes (A), until operators manually de-energized Transformer T1B-4A by opening Breaker 1A4-10. In accordance with 480 VAC, 4160 VAC and Fire protection system design criteria and IEEE Standards, a fault should be isolated by the breaker closest to the fault. This would have isolated and arrested the fault and prevented it from impacting other electrical buses. However, Breaker 1A4-10's breaker trip setpoint was such that a phase-to-phase fault on the line side of Breaker 1B4A would not be cleared. This allowed the fire to continue, produce combustion products, and develop the subsequent ground fault between the BT-1B4A breaker and the island bus. Although the licensee generated CR 2012-01630 on March 1, 2012, which acknowledged this condition, the licensee had yet to complete an analyze the adequacy of the breaker trip set points as of the conclusion of this inspection period. Therefore, the 4160 VAC bus is still not protected against an arc fault event on the 480 VAC bus upstream of the 480 VAC feeder breaker and this vulnerability is still present.

3) The bus separation scheme design was inadequate to meet the system's design criteria , IEEE standards, and the 1971 NRC Standard Review Plan (SRP). (Note: however, the inspectors recognize that OPPD was licensed to operate FCS prior to the SRP). OPPD's bus separation scheme design allowed combustion products from the Bus 1B4A fire to be communicated to and affect the island bus because of the physical configuration of the bus duct work and because there is only one bus tie breaker on each end of the island buses. This configuration and the fire event resulted in the development of an electrical short to ground between Breaker BT-1B4A and Island Bus 1B3A-4A, which was powered from the opposite safety bus (Bus 1B3A). Thus both independent trains of vital AC power were adversely affected by a fault on a single bus. FCS's corrective actions restored the original configuration of the 480V switchgear. The inspectors were not aware of any formal evaluation which reviewed this design vulnerability and/or operability evaluation as of the end of the inspection period. This design vulnerability is still present.

4) The DC bus separation scheme design for the DC buses was questioned by the inspectors. During the fire event on June 7, 2011, grounds developed on both DC buses. OPPD evaluated the inspector's concern and were able to demonstrate that there was no adverse impact on the DC buses. While the

condition did result in degradation of the DC busses,t he DC buses are an ungrounded system by design; therefore, a single ground would not impact system operation. This condition was determined not to be a SCAQ, because it is consistent with plant and system design basis and did not contribute to the unexpected plant response during the event. This concern was adequately evaluated by OPPD and can be considered closed.

5) The breaker coordination scheme design did not respond as expected during the fire event. Breaker 1B3A tripped when a fault developed on Island Bus 1B3A-4A, which resulted in both Bus 1B3A and Island Bus 1B3A-4A being lost during the event. In accordance with system design requirements, Breaker BT-1B3A should have isolated the fault. Because of the fire and breaker coordination failure, six of nine vital 480 V buses were either manually or automatically de-energized during the event, and minimum ECCS system capacity was not maintained. This condition was the focus of CR 2011-6621. Corrective actions developed were reviewed by the inspectors and determined to be adequate to preclude repetition of this SCAQ. This SCAQ and the associated finding 2012-004-04 can be closed

The inspectors determined the above conditions were SCAQ based upon the following. Condition #1 is a SCAQ because it resulted in the initiation of an electrical fault on a single 480 VAC bus and the resulting fire which caused significant equipment damage and resulted in an EAL declaration. Condition #2 is a SCAQ because it 1) allows a fault on a single 480 VAC bus to adversely impact the associated 4160 VAC bus and the remaining 480 VAC on this train, 2) prevents the fault from being deenergized thus allowing the fire to burn for an additional 42 seconds causing significant equipment damage, and 3) develops charged particles and soot which allow the design vulnerability discussed in condition 3 to be exploited. Condition #3 is a SCAQ because this design vulnerability is the mechanism which allows a single fault to impact both trains of safety related equipment (ECCS, 480VAC, and SSE) and thus is not with the design basis and is an unanalyzed condition. As a result, during the event, the fault impacted the 1B3A-1B4A island bus which is powered from the opposite train from the fault. Condition #5 is a SCAQ because it resulted in a loss of breaker coordination which is relied upon per the Fire Protection Safe Shutdown Design to protect both trains of SSE during a postulated fire event, thus this was beyond the system design basis and was an unanalyzed condition. As a result during the event, a second bus on the opposite train from the fault was lost and all high pressure make up water sources (HPSI and Charging) were lost and could not be restored via remote manual operator action.

During this inspection period the inspectors focused on SCAQs 2, and 3. In 1991, FCS completed a breaker coordination study of the 4160 VAC and 480 VAC distribution systems. The inspectors identified that the breaker coordination survey correctly identified that the breaker trip setpoints does not provide full protection against a 4160VAC/480VAC transformer fault or a 480V Load Center Bus Fault. The study evaluated that this was acceptable due to “the possibility of a fault is extremely small" and the fact that the USAR Section 8.3.1 does not state that all breakers are coordinated.

The inspectors challenged this conclusion on the basis that the major concern is fault protection and clearance. The June 7, 2011 fire demonstrated that not showing protection against a 480 VAC load center fault would prevent the fault from being cleared, allow a 480 VAC fault to adversely impact the associated 4160 VAC bus. This lack of protection turned a fire on a single 480VAC bus into an unisolable fault on the 4160 VAC bus, which is the worst case design basis single failure for ECCS since the entire 4160 VAC bus and thus each of the associated 480VAC on the train are also lost. In addition, since the fire is allowed to continue to burn, the known bus separation design vulnerability is allowed to be exploited as soot and charged particle are allowed to collect on the associated bus tie breaker (physically located in the same switchgear cabinet) and develop a fault on the opposite train. This takes plant response outside of the design basis. Thus, the inspectors questioned the validity of the study's conclusion. This concern was still under review by the inspectors at the end of the inspection period.

(3) Findings

.1 Untimely Corrective Actions for 480 VAC Breaker Issues

Introduction: The inspectors identified a Green noncited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions." Specifically, FCS failed to take timely corrective actions to address non conforming conditions identified in several breakers during their review of the June 7, 2011, 1B4A Bus fire and abnormal system response event. Specifically, several breakers were observed to have significant breaker cradle assembly to bus stab misalignment and high impedance connections were identified. These conditions were determined to have been the cause of the 1B4A bus bar failure that initiated the fire; however, this condition was not corrected for several months. Additionally, FCS continued to operate with a degraded 1B3A breaker for nine additional months after the breaker tripped unexpectedly during the event. The breaker remained in service until February 2012.

Description: Following the June 7, 2011, 1B4A breaker fire and abnormal 480 VAC system response event, FCS wrote several CRs and conducted Root Cause Analysis (RCA) CR 2011-05414 related to the why the fire occurred. A separate CR (2011-6621) was written to document and review the unexpected tripping of the 1B3A breaker during the fire event. FCS initially failed to properly evaluate the significance of the event as an event significant to nuclear safety, in accordance with FCS corrective action program procedure . FCS only evaluated the event with respect to the plant conditions at the time of the fire (i.e., the plant was shutdown for a refueling outage). In September 2011, during the NRC's Special Inspection of the fire event, FCS revised its risk assessment and determined that if the fire had occurred at power, it would have been an event significant to nuclear safety because of the loss of all high pressure injection sources (HPSI and charging pumps), that adversely impacted both trains of safe shutdown equipment (SSE).

In July 2011, boroscope inspections of the undamaged 480 VAC breakers were conducted as an Extent of Condition (EOC) review. These inspections revealed that four other breakers had significant breaker cradle finger to bus stab misalignment and appeared to be contacting the stabs beyond the silver plating on the stabs and created a high impedance connection. This was the same failure mechanism that CR 2011-5414 concluded was the most likely cause of the 1B4A bus failure and fire. However, once this condition adverse to quality was identified to exist, the condition was not corrected until November 2011. During this work, one of the breakers, Breaker 1B3C was found to have discolorations on the fingers of the cradle assembly, and this was believed to be heat related. This indicated that at least one other breaker was potentially progressing down the same failure mechanism as 1B4A.

As discussed in IR 05000285/2011014, the initiating event likelihood of a fire was calculated to increase to 7.0 x 10-2/year from a baseline likelihood of 2.5 x 10-5 / year due to this misalignment condition. This significant increase in the likelihood of another fire occurring due to the same cause as the June 7th, 2011, fire, would make the misalignment identified in July 2011 a Significant Condition Adverse to Quality (SCAQ). SCAQs and CAQs must be identified and corrected commensurate with the safety significance of the issue. Because this issue was determined to have Red (i.e., high) safety significantce by the NRC and to be significant to nuclear safety in accordance with FCS's own risk re-assessment in September 2011, the NRC determined that waiting to address the issue until November 2011 was not determined to be a timely corrective action.

In addition, during the 1B4A breaker fire, the 1B3A breaker tripped unexpectedly to clear a fault induced at the BT-1B4A Bus Tie breaker. This fault should have been cleared by the BT-1B3A breaker, but the 1B3A breaker tripped first, contrary to the FCS Breaker Coordination design Scheme. CR 2011-6621 was written to document the abnormal 480 VAC system response, but it was classified originally as a 'C' level CR, and no formal evaluation was assigned. It was determined via CR 2011-5514 that 1B3A tripped before BT-1B3A because the breaker coordination curves were set close to each other in a manner that allowed 1B3A to trip first (i.e., it "won the relay race"). The 1B3A breaker was returned to service on June 22, 2011. In September 2011, this CR was brought back to the station's corrective action review board (CARB) meeting and reassigned as a level 'A' CR, and a root cause evaluation was assigned. This was due to the fire event risk being re-evaluated. This root cause rejected the "relay race" explanation and identified several potential failure mechanisms that could have caused the breaker to trip outside the coordination scheme. Troubleshooting was commenced by testing the breakers locally and by testing the BT-1B3A and 1B3A breakers at the NLI facility in October 2011. These tests did not identify any problems, and the breakers were returned to FCS and returned to service.

In February 2012, both breakers BT-1B3A and 1B3A were removed and sent to NLI's factory test facility. During this round of testing, the symptoms observed on June 7, 2011, were repeated. With 1B3A and BT-1B3A in series with a fault source, the 1B3A breaker tripped instantaneously before the BT-1B3A breaker could trip. Further inspection revealed that the WAGO jumpers disabling the Zone Selective Interlock (ZSI) function were not installed in the correct location; therefore, the ZSI feature was not disabled as originally intended. The jumpers were restored to their proper positions, and proper breaker coordination was observed. Licensee inspections were conducted on the remaining breakers at FCS in a timely manner, and no further issues were identified.

However, the fact that the 1B3A breaker was in service in a known degraded condition from June 22, 2011, until February 27, 2012, is another example of corrective actions not being timely, and exposing the plant to unnecessary risk.

Analysis: The failure to take timely corrective actions for known SCAQs or CAQs was within FCS's ability to foresee and prevent and is therefore a performance deficiency. The performance deficiency was evaluated using NRC Inspection Manual Chapter 0612, Appendix B, "Issue Screening," and the issue was determined to be more than minor because it affected the Initiating Events Cornerstone attribute of protection against external events (i.e., fire) because the condition that contributed to the fire event was left uncorrected. The issue adversely affected the associated cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. This issue also affected the Mitigating System Cornerstone.

The inspectors evaluated the finding using NRC Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings." IMC 0609, Attachment 4 directs the user to use of IMC 0609, Appendix G, "Shutdown Operations SDP," since FCS was in cold shutdown during the entire exposure period. Using IMC 0609 Appendix G, Attachment 1, "Phase 1 Operational Checklists for Both PWRs and BWRs," Checklist 4, "PWR Refueling Operation: RCS level > 23' or PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer," the issue screens to having Green (i.e., very low) safety significance because RCS makeup capability was not degraded since one or more low pressure makeup water sources would remain available. The inspectors determined that the issue had a cross-cutting aspect in the area of Human Performance, Decision Making, in that the licensee failed to use conservative assumptions and 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 (H.1.b).

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," states that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, from July 2011 until November 2011, a SCAQ related to the misalignment of the 480 VAC breakers, which was identified as the cause of the 1B3A fire, was identified to exist in other equipment, but corrective actions to preclude repetition were not taken in a timely manner. Additionally from June 22, 2011, until February 27, 2012, a known degraded breaker was allowed to remain in service for approximately 8 additional months until the cause was identified. This was also not timely. FCS corrected the nonconforming conditions in the breakers and has revised its corrective action program. However, because this violation was of very low safety significance, and FCS has entered this issue into their its corrective action program as CRs 2012-01884 and 2011-5414, the NRC is treating this as an NCV in accordance with Section 2.3.2 of the NRC Enforcement Policy; 05000285/2012005-02, "Untimely Corrective Actions for 480 VAC Breaker Issues."

.e Third-Party Safety Culture Assessment

Item 1.e is included in the restart checklist because the NRC recognizes the importance of nuclear plant licensees establishing and maintaining a strong safety culture, a work environment where management and employees are dedicated to putting safety first. In addition, nuclear power plants should have a work environment where employees are encouraged to raise safety concerns, and where concerns are promptly reviewed, given the proper priority based on their potential safety significance, and appropriately resolved with timely feedback to the originator of the concerns and to other employees.

(1) Inspection Scope

The NRC attended safety conscious work environment (SCWE) training that FCS provided to its supervisors on September 13, 2012. The site vice president introduced the training class with a discussion of why the training was being conducted. The training material included the topics of employee protection, regulations, the definition of SCWE and its relationship to safety culture, the attributes of SCWE, discrimination, and the safety culture survey results performed at FCS in May 2012. The instructor emphasized the importance of encouraging people to enter issues into the FCS corrective action program (CAP).

(2) Assessment

Inspectors thought that the training content was adequate and that the opportunity at the end of the training for supervisors to discuss the safety culture survey results was beneficial. During the training, the instructor discussed the differences between the concepts of perception of retaliation versus proof of retaliation. A question of perception versus proof of retaliation also came up during the meeting on July 19, 2012, in which fundamental performance deficiencies were discussed. NRC inspectors commented to FCS staff during a weekly debrief that insights from the site's employee concerns program manager, union, and human resources department could have been incorporated into the training to help clarify what types of behavior FCS employees are perceiving as retaliation.

(3) Findings

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

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

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

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

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

On August 30, 2011, Fort Calhoun Station issued Revision 1 to the "Fort Calhoun Station Post-Flooding Recovery Action Plan," (FRAP) that provided for extensive reviews of plant systems, structures, and components to assess the impact of the floodwaters. On September 2, 2011, the NRC issued Confirmatory Action Letter (CAL) 4-11-003, listing 235 items described in the Fort Calhoun Station Post-Flooding Recovery Action Plan that the licensee committed to complete. These 235 items were broken down into three sections: items to complete prior to exceeding 210 degrees Fahrenheit in the reactor coolant system, items to complete prior to reactor criticality; and items to complete following restart of the plant. On June 11, 2012, the NRC issued CAL 4-12-002. This CAL incorporates all the actions required by CAL 4-11-003.

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

(1) <u>CAL Action Item 1.2.1.4</u>

i. <u>Inspection Scope</u>

The purpose of Action Item 1.2.1.4 was to return B.5.b materials to their proper location. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

During the 2011 flood some B.5.b materials were displaced from their normal location in the FCS warehouse to other locations on site.

After flood waters receded, the B.5.b materials were relocated from their temporary location in the training center truck bay to their permanent location. The licensee inventoried the equipment per Attachment 11 of OCAG-1, “Operational Contingency Action Guideline.”

The inspectors performed an independent inventory of all B.5.b materials listed in attachments 8, 9, 10, and 11 to ensure all equipment was accounted for.

This activity constitutes completion of Action Item 1.2.1.4 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(2) CAL Action Item 2.1.1.3

i. Inspection Scope

The purpose of Action Item 2.1.1.3 was to flush fire protection piping connected to the fire protection header ring which flowed river water during flood mitigation actions. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

In preparation for the Missouri River flooding in 2011, the licensee installed a water filled protection device around the plant. This required large quantities of water to be provided for extended periods of time. The licensee utilized the electric driven fire pump, PF-1A to fill the individual sections of the protection device with some of the exterior fire hose cabinets. This activity deposited river water into the underground fire main piping around the plant.

The licensee performed OP-ST-FP-0011, “Fire Protection System Hose Station Operability Test” to flush the underground fire main and to operate all exterior fire hydrants. Each fire hydrant was flushed for approximately 2 hours with clean, fresh water.

The inspectors reviewed the USAR, procedure requirements, and technical specifications to ensure that the surveillance activities demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. During performance of the surveillance test, Fire Hydrant FP-3C was identified as degraded during flushing activities.

The licensee created a new action item in the flood recovery action plan, 2.1.3.8, which was to replace FP-3C. The licensee replaced the fire hydrant, as well as its associated isolation valve, FP-114.

The inspectors observed the installation of the fire hydrant and isolation valve, as well as the postmaintenance testing to ensure the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed, and acceptance criteria were clear and demonstrated operational readiness; and test instrumentation was appropriate.

This activity constitutes completion of Action Item 2.1.1.3 as described in Confirmatory Action Letter 4-12-002, as well as flood recovery plan action item 2.1.3.8.

Findings

ii. No findings were identified.

(3) CAL Action Item 2.1.1.9

i. Inspection Scope

The purpose of Action Item 2.1.1.9 was to complete full flow testing of fire pumps. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

Due to the flooding conditions and the location of the water filled protection device, access to the fire protection test header valves and general area required for testing equipment was restricted. To complete full flow testing of the fire pumps, a test rig was installed on a truck with fire hoses attached to the test header. This test configuration required the areas west of the intake structure to be clear. Due to the conditions from the flooding event the tests had to be delayed until the flood waters had receded and the water filled protection device was removed.

Surveillance Tests SE-ST-FP-0002 'Fire Protection System Motor Driven Fire Pump Full Flow Test' and ST SE-STFP-0003 'Fire Protection System Diesel Driven Fire Pump Full Flow Test' were completed as soon as the testing area and equipment was accessible. Both the electric motor driven fire pump FP-1A and the diesel driven fire pump FP-1 B passed their respective surveillance tests and were returned to service and declared operable.

The inspectors reviewed the USAR, procedure requirements, and technical specifications to ensure that the surveillance activities demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions.

The inspectors witnessed and reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant

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

This activity constitutes completion of Action Item 1.2.1.4 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(4) CAL Action Item 2.2.1.32

i. Inspection Scope

The purpose of Action Item 2.2.1.32 was to assess the effects of the flood on the Communications 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, any outstanding preventive or corrective maintenance required, and reviewed all open condition reports, as well as all condition reports created since January 1, 2011. The inspectors also conducted a complete system walkdown to identify any adverse conditions and to verify all system components were functioning properly. The inspectors compared the results of their independent assessment to those contained in the licensee's "Flooding Recovery Startup System Health Assessment" report.

The Plant Communications System uses a combination of dial telephones, dedicated telephone lines, intra-plant intercom/paging facilities, Paging System and a 800 MHz Radio Communication System for on-site information relaying and alarm notification. It also provides off-site communications with other facilities and support personnel. The Plant Communications System does not perform any safety related functions.

The inspectors identified no adverse conditions associated with the Plant Communications System.

This activity constitutes completion of Action Item 2.2.1.32 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(5) CAL Action Item 2.3.1.1

i. Inspection Scope

The purpose of Action Item 2.3.1.1 was to assess whether motors were to be tested for possible use, refurbished or replaced. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The licensee determined that five normally dry pump motors were wetted for some period of time: the three circulating water pump motors, CW-1A, CW-1B, and CW-1C; and the Demineralized Water Storage Tank inlet and outlet pump motors, DW-69 and DW-70.

The inspectors performed an independent assessment of which motors may have been wetted. The assessment included individual inspector walkdowns during and after the flood, and conversations with inspectors who had been present during the site flooding in 2011. The inspectors also searched all opened condition reports since the onset of flooding and concurred with the licensee that the circulating water and demineralized water pumps were the only normally dry pump motors that had been wetted by floodwaters.

The licensee also created action items 2.3.1.2 through 2.3.1.8 to track completion of items associated with the circulating water pump motors, and action items

2.3.1.9 through 2.3.1.16 to track completion of items associated with the demineralized water pump motors.

In addition to these five pumps, the switchgear room ventilation condensing units, VA-89 and VA-90 were flooded when the water filled protection device around the plant collapsed. These condensing units were repaired prior to the issuance of the flood recovery plan.

This activity constitutes completion of Action Item 2.3.1.1 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(6) CAL Action Items 2.3.1.2 and 2.3.1.1

i. Inspection Scope

The purpose of Action Items 2.3.1.2 and 2.3.1.3 were to take oil sample from bearing housings and evaluate if water had gotten in contact with the bearings in the circulating water pump motors, CW-1A, CW-1B, and CQ-1C. These items were required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The licensee took cold oil samples from each of the circulating water pump motor's upper and lower bearing reservoirs on September 6 and 7, 2011. The samples were then sent to a third party laboratory for analysis.

The inspectors observed the sampling of the oil and reviewed the analyses results. Each of the 6 samples were first tested for water contamination via the Crackel Test. The Crackle Test is a standard laboratory test to detect the presence of water in lubricating oil. A drop of oil is placed on a hotplate that has been heated to approximately 400 degrees Fahrenheit. The sample then bubbles, spits, crackles or pops when moisture is present. The Crackel test showed undetectable for water content for all samples except for the inboard bearing reservoir for CW-1C motor.

The samples were then tested utilizing the Karl Fisher Titration method. This method uses anode and titrant solutions to determine concentrations of water in the oil. The Karl Fisher Titration results showed the inboard bearing reservoir for CW-1C motor to contain approximately 110 parts per million (ppm) water, where the other five bearing reservoirs contained between 20.0 and 21.5 ppm water. This is indicative of flood water ingress into the inboard bearing reservoir for CW-1C motor.

The refurbishment of CW-1C pump motor will be completed and evaluated under Confirmatory Action Letter item 2.3.1.4, This activity constitutes completion of Action Items 2.3.1.2 and 2.3.1.3 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(7) CAL Action Items 2.3.1.5 and 2.3.1.6

i. Inspection Scope

The purpose of Action Items 2.3.1.5 and 2.3.1.6 were to perform visual and boroscope inspections of the circulating water pump motors and evaluate the results. These items were required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The licensee performed a visual inspection of the circulating water pump motor internals and termination boxes on September 8, 2011.

The inspectors performed an independent visual inspection of the pump motors, and observed the licensee using the boroscope. The inspectors evaluated the boroscope photographs and compared them to their visual inspection.

The inspection showed no signs of debris, silt, moisture or corrosion. The motors did contain a fine film of dust throughout the stator winding as a normal result of operation. The inspection showed similar results for all three motors. A cleaner area was observed near the termination box opening into the motor of the CW-1C motor. This was indicative of where water entered the motor.

No abnormal degradation was noted in any of the three pump motors. The refurbishment of CW-1C pump motor will be completed as a result of water intrusion into the motor oil as described in action items 2.3.1.2 and 2.3.1.3 and evaluated under Confirmatory Action Letter item 2.3.1.4,

This activity constitutes completion of Action Items 2.3.1.5 and 2.3.1.6 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(8) CAL Action Item 3.2.1.2

i. Inspection Scope

The purpose of Action Item 3.2.1.2 was to test maintenance rule low voltage power cable on cables which had been subjected to wetting/submergence. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The licensee performed megger testing on low voltage (480 volt) cable in November 2011. The population of cables was those which were exposed to water, traversing through manholes 5 and 31: the feeder cables for motor control centers MCC-3B3 and MCC-4C4.

The inspectors observed the licensee's megger testing and analyzed the result. A megger test is performed to ensure the adequacy of the insulation in a cable. In 480 volt cables, 500 volts are applied to the cable for one minute, and the resistance is measured. The acceptance criteria for 480 volt cables is 1.48 megohms. The inspectors verified that the resistance on the cables for MCC-3B3 were greater than 50,000 megohms, and for MCC-4C4 were greater than 2,000 megohms.

This activity constitutes completion of Action Item 3.2.1.2 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

(9) CAL Action Item 3.2.1.3

i. Inspection Scope

The purpose of Action Item 3.2.1.3 was to test maintenance rule low voltage control and instrumentation on cables which had been subjected to wetting/submergence. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The licensee performed megger testing on low voltage instrumentation and control cables in October, 2011. The population of cables was those which were exposed to water, traversing through manholes 5 and 31: motor driven fire pump, FP-1A, control cable; the four raw water pump discharge valve control cables, HCV-2850, HCV-2851, HCV-2852, and HCV-2853; and the six raw water discharge header isolation valve control cables, HCV-2874A & B, HCV-2875A & B, and HCV-2876A & B.

The inspectors observed the licensee's megger testing and analyzed the result. A megger test is performed to ensure the adequacy of the insulation in a cable. In low voltage control and instrumentation cables, 250 volts are applied to the cable for one minute, and the resistance is measured. The acceptance criteria for these cables is 1.13 megohms. The inspectors verified that the resistance on all of the cables was greater than 2,000 megohms.

This activity constitutes completion of Action Item 3.2.1.3 as described in Confirmatory Action Letter 4-12-002.

ii. Findings

No findings were identified.

.3 Adequacy of Significant Programs and Processes

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

.a Corrective Action Program

(1) Inspection Scope

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

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

For the assessment period covered by this inspection report, the onsite activities included the observation of CAP meetings such as the Department Station Corrective Action Review Board (DCARB), which was observed for the Operations Department, and a presentation of the licensee's corrective actions taken to date. The presentation also included an explanation of the root causes identified as a result of the licensee's review of the CAP and what the next steps are for their improvement plan. In addition, the inspectors interviewed site personnel associated with the Performance Improvement department to continue to get a better understanding of the site CAP processes. The in-office activities, which were conducted at the inspectors' regular duty stations, consisted of reviews of root cause analyses and procedures associated with the Corrective Action Program.

(2) Assessment

During this assessment period, the inspectors attended one DCARB meeting for the Operations Department. To be able to reasonably assess these processes, the inspectors will continue to attend more of these meetings and observe more of the CAP processes during future on-site inspection weeks. In general, the inspectors noted a general attitude to follow the CAP procedures and healthy willingness to express dissenting views during CAP meetings. However, during the course of interviews, plant tours and interactions with plant personnel, the inspectors have also noted a general behavioral issue with the threshold to initiating Condition Reports (CRs). The inspectors have noted that, especially with lower level issues, the workers opt for an attempt to repair the condition in-place and not writing a condition report to document the deficiency, and place it in the CAP. The inspectors noted that this approach could prevent issues from being placed in the CAP at an early stage.

(3) Findings

No findings of significance were identified.

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

(1) Inspection Scope

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

(2) Assessment

During the inspection period, OPPD completed the discovery phase of its evaluations of this issue. The discovery phase was designed to identify all work orders (WOs) where non safety related parts were issued for jobs involving safety-related SSCs. This process identified 2100 WOs to be evaluated to determine if non safety related parts were installed in safety-related systems and, if so, whether these parts impacted the systems' functionality and operability. At the end of the inspection period, the licensee had reviewed approximately 40 percent of the 2100 WOs, and approximately 15 of those WOs required an evaluation of the impact on system functionality and operability. The NRC inspectors will continue to review all instances of WO issues that resulted in system functionality evaluations, and the team will assess a sampling of the WOs for which further evaluations were performed to determine the effectiveness of the licensee's review. This restart checklist item will remain open until all WOs have been screened and questions related to operability of SSCs required for Modes 1 and 2 have been appropriately evaluated and addressed.

During the inspection period, FCS changed its scope expansion criteria for this project. Originally, FCS's scope expansion was based on criteria related to the number of WOs discovered in the discovery phase and adding more WOs to the population to be evaluated. FCS changed this scope expansion criteria to one based upon components evaluated to have been installed in a safety related application and requiring further review. When an item is found to meet this criteria, the scope is expanded to search for additional WOs where this part was issued beyond the original 5-year scope and in other systems. The change was made to allow the scope expansion to be more risk based. The inspectors will assess if the revised scope expansion criteria is as effective as identifying vulnerabilities which occurred beyond the original 5-year scope.

In addition, FCS is in the process of replacing its "CQE/non-CQE" terminology with "safety-related/non-safety related" terminology. FCS staff expects the updates to station programs and procedures to be completed in 2012. The "CQE-to-safety related" terminology conversion is expected to be completed in October 2013, and the "non-CQE-to-non-safety related" terminology conversion is supposed to be completed by January 2014.

(3) Findings

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

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

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 HELB effects. In preparations for a postponed extended power up-rate, Fort Calhoun Station reviewed HELB calculations. FCS personnel found that it was lacking adequate documentation and calculations for HELB effects in some areas. The restart checklist includes an NRC assessment of FCS's HELB analyses and documents to ensure the plant is within its licensing and design basis for HELB effects. The NRC will also assess the licensee's qualifications and documentation for certifying equipment for harsh environments.

(1) Inspection Scope

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

(2) Assessment

During this inspection period, OPPD continued to evaluate concerns related to containment electrical penetrations discussed in LER 2852012002, and an overall review of the Environmental Qualification program and HELB program including a reassessment of the HARSH environment files and program scope and basis. These reviews were still in progress at the end of the inspection period. Therefore, the NRC's review of this restart checklist item is still in progress. Closure of this restart checklist item will be dependent on, in part, the evaluation and resolution of the issues discussed in the aforementioned LERs, including any operability concerns.

(3) Findings

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

.c Design Changes and Modifications

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

This item assesses the effectiveness of the licensee's implementation of changes to facility structures, systems, and components, risk significant normal and emergency operating procedures, test programs, evaluations required by 10 CFR 50.59, and the updated final safety analysis report. The NRC will inspect to provide assurance that changes have been appropriately implemented.

.i Vendor Modification Control

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

(1) Inspection Scope

NRC inspectors interviewed station personnel and contractors that performed the reviews. Inspectors also reviewed the collective evaluation condition report.

(2) Assessment

The licensee completed its latest version of the collective evaluation condition report, which summarized the results of its review of modification packages prepared by vendors. The condition report mentions that significant issues were identified; however, the licensee stated that it does not plan to perform a root cause analysis on this topic. The inspector discussed some discrepancies in the report in the characterization of identified issues. For example, the overall conclusion in the report was that the identified issues were administrative; however, another section of the report mentions significant issues were identified. The licensee stated that it was aware of the discrepancies and is revising the report. Inspectors also expressed a concern that the condition report stated what the causal analysis should conclude instead of allowing the causal analysis process to come to its own conclusions. The inspectors expressed the concern that the effort was potentially being biased by the results of the organizational effectiveness root cause analysis. The licensee stated that a final report for this issue was still in progress.

(3) Findings

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

.ii 10 CFR 50.59 Screening and Safety Evaluations

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

(1) Inspection Scope

NRC inspectors interviewed station personnel and contractors that performed the reviews. Inspectors reviewed a sample of the condition reports generated from the review.

(2) Assessment

The licensee stated that they completed their review of 50.59 documents and the collective evaluation condition report and are in the stage of developing a final report before commencing a root cause analysis for the identified issues. The collective evaluation condition report summarizes the results of the licensee's review of 50.59 documents. The report attachment contained the same statement as the vendor modification report about what the cause analysis should state, which further supported the inspectors' concern that the organization effectiveness root cause analysis results could bias other root cause analyses. The licensee stated that they would remove these statements from the condition reports.

The condition reports documenting the results of each 50.59 document review contained due dates for when the identified issues would be corrected, if FCS staff decided the issues had to be corrected. The inspectors noticed that some of the corrective actions (e.g., updating the 50.59 documents with applicable design basis information to support the conclusions) were deferred until after restart. Inspectors also noticed a condition report that identified that design basis information was not adequately incorporated or referenced in a 50.59 for an engineering change (EC); however, FCS staff responded to the condition report that there was no benefit to correcting the EC package because the summary of the modification was already sent to the NRC, and the result of the 50.59 would have been the same. The licensee stated that the contractors performing the reviews were relying on experience and judgment to gauge whether NRC approval would have been needed for determining the due dates for correcting issues.

NRC inspectors attended a portion of 50.59 training that was provided by a contractor to FCS staff. The training was thorough and of high quality.

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

(3) Findings

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

.d Maintenance Programs

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

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

The NRC will assess the licensee's maintenance programs, including preventative maintenance, compliance with vendor recommendations, post-maintenance testing programs, and establishing and controlling equipment service life.

(1) Inspection Scope

.i Vendor Manuals and Vendor Informational Control Programs

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

.ii Equipment Service Life

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

(2) Assessment

The team noted that a new apparent cause analysis is being performed for the vendor manual area and the targeted completion date for this new analysis is October 30, 2012. The licensee does not currently have a schedule for completion of all corrective actions. The NRC will close out this issue for restart after the inspections verify that the station has 1) completed the new apparent cause analysis, 2) completed the corrective actions from the apparent cause analysis, and 3) completed all actions necessary to prevent re-occurrence. Additionally, because the vendor manuals contain the service life requirements for most equipment and their subcomponents, the inspectors will need to complete the service life corrective actions as well to ensure proper reconciliation of these programs was accomplished. The licensee wrote Condition Report CR 2012-09215 to address the reconciliation issue.

The team noted that the root cause analysis being performed for the equipment service life issue is scheduled to be completed on November 28, 2012. The licensee does not currently have a schedule for completion of all corrective actions. The NRC will close out this issue for restart after the inspections verify that the station has 1) completed the root cause analysis, 2) completed the corrective actions from the root cause analysis, and 3) completed all actions necessary to prevent re-occurrence. As mentioned above, proper reconciliation will need to be verified for this issue as well (Condition Report CR 2012-09215).

(3) Findings

No Findings of significance were identified.

.e Operability Process

Improper evaluations of degraded and/or non-conforming conditions may result in continued operation with a structure, system, or component that is not capable of performing its design function.

(1) Inspection Scope

.i Operability Determination Process

NRC inspections determined that Fort Calhoun Station did not consistently conduct adequate Operability Evaluations to ensure that the impacts of degraded conditions on plant operations are fully understood. In response, the licensee will assess their operability evaluation program and develop corrective actions to improve performance. The NRC will assess the licensee's operability determination process reviews. The NRC will inspect a sample of operability determinations to ensure proper implementation of the licensee's process and ensure evaluations were correct.

.ii Degraded and Non-conforming Conditions

NRC inspection determined that some equipment identified as "operable but degraded" remained degraded until subsequent failure occurred. Fort Calhoun Station processes did not adequately identify degraded equipment or restore equipment from a degraded condition to full qualifications in a timely manner. In response, Fort Calhoun Station will assess their controls for the review of "operable but degraded" equipment. The NRC will evaluate the effectiveness of the changes made to the licensee's tracking and treatment of "operable but degraded" equipment.

(2) Assessment

The team noted that the root cause analysis being performed for the two issues of operability and degraded and non-conforming conditions is scheduled to be completed on November 16, 2012. The licensee does not currently have a schedule for completion of all corrective actions. The NRC will close out these two issues for restart after the inspections verify that the station has 1) completed the root cause analysis, 2) completed the corrective actions from the root cause analysis, and 3) completed all actions necessary to prevent re-occurrence.

The team reviewed several condition reports related to issues with containment spray pumps, component cooling water pumps, and the emergency diesel generators. The team also performed interviews and attended discussions with the licensee regarding the operating experience program. A detailed write up on these systems, components, and programs will be completed during the next six-week inspection period.

(3) Findings

No Findings of significance were identified.

4OA5 Other Inspection Activities (TI 2515/188)

.1 <u>(Opened and Closed)</u> <u>Temporary Instruction 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns</u>

NRC inspectors performed inspection activities to independently verify that Fort Calhoun Station conducted seismic walkdown activities using an NRC-endorsed seismic walkdown methodology. The seismic walkdowns are being performed at all sites in response to Enclosure 3 of 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).

(1) Inspection Scope

The inspectors accompanied the licensee on their seismic walkdowns of :

- The Motor Driven Auxiliary Feedwater Pump and associate equipment in Room 19 on August 8, 2012.
- The 'B' EDG and associated equipment on August14, 2012.

- Walk down of the 1A2/1A4 switchgear area and inspection of the 1A4-9 breaker cubicle.

The inspectors verified that the licensee confirmed that the following seismic features associated with the above equipment and systems were free of potential adverse seismic conditions by verifying:

- Anchorages were free of bent, broken, missing or loose hardware.
- Anchorages were free of corrosion that is more than mild surface oxidation.
- Anchorages were free of visible cracks in the concrete near the anchors.
- Anchorage configurations were consistent with plant documentation.
- SSCs will not be damaged from impact by nearby equipment or structures.
- Overhead equipment, distribution systems, ceiling tiles and lighting, and masonry block walls are secure and not likely to collapse onto the equipment.
- Attached lines have adequate flexibility to avoid damage.
- The area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area.
- The area appears to be free of potentially adverse seismic interactions that could cause a fire in the area.
- The area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

The inspectors independently performed their walkdown and verified that the following areas were inspected and seismic features verified:

- Walkdown of the Component Cooling Water Pump Area and associated equipment on September 18, 2012.
- Walk by of the Charging Pump Room and associated equipment on September 19, 2012.

Additionally, inspectors verified that items that could allow the spent fuel pool to drain down rapidly were added to the SWEL and these items were walked down by the licensee.

(2) Findings and Observations:

No NRC-identified or self-revealing findings were identified.

The walkdowns were performed by contract personnel with support from OPPD's operations and security departments. FCS appropriately conducted the walkdowns in accordance with the industry guidance. Observations were documented in the corrective action program as condition reports, as appropriate. The inspectors observed that FCS completed walkdowns of all accessable equipment. For equipment which was inaccessible (such as energized electrical busses) or equipment for which full walkdowns could not be completed, the walkdowns were documented as not completed and followup inspections were scheduled for system outage windows. The inspection of these items is expected to be completed prior to plant restart.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 18, 2012, the inspectors presented the inspection results to Mr. Mike Prospero, Plant Manager, and other members of the licensee staff. Additionally, on November 7, 2012 one finding was recharacterized as an Severity Level IV, cited violation. 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

05000285-2011-010-00	LER	Fire Causes a Circuit Breaker to Open Outside Design Assumptions
05000285-2012-014-00	LER	Containment Beam 22 Loading Conditions outside of the Allowable Limits
05000285-2012-015-00	LER	Electrical Equipment Impacted by High Energy Line Break Outside of Containment
05000285-2012-016-00	LER	Unanalyzed Charging System Socket Welds to the Reactor Coolant System
05000285/2012005-01	NOV	Failure to Update the Safety Analysis Report – Solid Waste

Opened and Closed

05000285/2012005-02	NCV	Untimely Corrective Actions for 480 VAC Breaker Issues
2515/188	TI	Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

LIST OF DOCUMENTS REVIEWED

Section 1R08: Inservice Inspection Activities

CONDITION REPORTS

2012-01123

PROCEDURES

NUMBER	TITLE	REVISION / DATE
CFTC-09-108	Field Service Report Steam Generator Secondary Side Services 2009 Outage	December 18, 2009
MRS-SSP-2229-CFTC1	Analysis of Eddy Current Data	0

DRAWINGS

NUMBER	TITLE	REVISION / DATE
E-925-096	Primary Piping Layout (Plan View)	18

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
CFTC-09-108	Field Service Report Steam Generator Secondary Side Services 2009 Outage	December 18, 2009
89361	Steam Generator Services April 2008 Refueling Outage	May 5, 2008
	Steam Generator Eddy Current Test Report – 2008 Refueling Outage	December 9, 2008
	Revised License Amendment Request, “Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process and Deletion of Sleeving as a Steam Generator Tube Repair Method”	August 30, 2006
FC06968	FCS RSG – Evaluation for the Impact of the RSG on FCS	0
EC31589	Replacement Steam Generators (Component)	0

	CFTC1_SG-B_20080401_ADI-ADH-CALLS	9/5/2012
	CFTC1_SG-A_20080401_ADI-ADH-CALLS	9/5/2012
	MHI Presentation on FCS Steam Generators	9/5/2012
	Page 11 from Final ECTReport08 R1 1364	4/2008
	CFTC1_SGA_Pri_sec_res_abs_channels_minus-ndf	9/25/2012
	CFTC1_SGB_Pri_sec_res_abs_channels_minus-ndf	9/25/2012
LTR-AMER-MKG-12-1715	Westinghouse Steam Generator Operational Assessment Scope	1
	Tubes looked at:SG11HCALROD00005 65 tubes	4/27/2008
	SG21HCALROD00002 181 tubes	4/30/2008

Section 1R15: Operability Evaluations

CONDITION REPORTS

2012-00550	2012-00551	2012-00552	2012-00657	2012-07085
2012-07143	2012-11933			

PROCEDURES

NUMBER	TITLE	REVISION / DATE
OP-12	Fueling Operations	64

DRAWINGS

NUMBER	TITLE	REVISION / DATE
11405-S-17	Reactor Plant Basement Floor Plan El 994’-0” Outline	17
11405-S-18	Reactor Plant Ground Floor Plan El 1013’-0” Outline	4
11405-S-19	Reactor Plant Operating Fl Plan El 1045’-0” and 1060’-0” Outline	15
11405-S-20	Reactor Plant Reactor Foundation and Fuel Pit – Sheet 1	2
11405-S-23	Reactor Plant Section & Details Outline – Sheet 2	5
11405-S-24	Reactor Plant Section & Details Outline – Sheet 3	4
11405-S-39	Reactor Plant Ground Fl Plan El 1013’-0” Reinf – Sheet 1	5

11405-S-41	Reactor Plant Operating Fl Plan El 1045’-0” and 1060’-0” Reinforcement – Sheet 1	4
11405-S-43	Reactor Plant Reactor Foundation & Fuel Pit Reinforcement – Sheet 1	2
11405-S-44	Reactor Plant Reactor Foundation & Fuel Pit Reinforcement – Sheet 2	2
11405-S-49	Auxiliary Building Misc Details	1
E-57	Refueling Area Crane Rail, Angle, Frame, Containment Plan 1038 Fy 6 In	1

CALCULATIONS

NUMBER	TITLE	REVISION / DATE
FC01420	Reactor Plant Operating Floor Design	0
FC03230	Containment Structural Design: Columns, Beams, Reinforcement, Various Elevations – Construction	2
FC06916	Seismic Analysis Calculation for the ReFueling Machine (FH-1)	0
FC06971	Past Operability Evaluation: RV Head Laydown Area Seismic Analysis	1
FC07176	Assessment of Concrete Beams at Elev. 1045'-0" in Containment for Rx Vessel Head Load	2

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
USAR 5.11	Structures Other Than Containment	10
USAR App F	Classification of Structures and Equipment and Seismic Criteria	9

Section 1R22: Surveillance Testing

WORK ORDERS

436013 436014 436015

PROCEDURES

NUMBER	TITLE	REVISION / DATE
OP-ST-FH-0001	Refueling System Fuel Handling Machine (FH-1) Interlocks Test	33
OP-ST-FH-0002	Refueling System Fuel Transfer System Interlocks Test	25
OP-ST-FH-0005	Refueling System Spent Fuel Handling Machine Refueling Interlocks Test	28

Section 4OA4: IMC 0350 Inspection Activities

CONDITION REPORTS (CR)

2011-8955	2011-8950	2011-8957	2011-7319	2011-8956
2011-5718	2011-5831	2011-5930	2011-5963	2011-5830
2011-5834	2012-12612	2012-13491	2011-2865	2011-6726
2011-7675	2011-5433	2011-8109	2012-03734	2001-02933
2011-09384	2012-09795	2005-01815	2008-05695	2010-06905
2011-00814	2012-02063	2012-03886	2012-04299	2012-04973
2012-09865	2012-09771	2012-09865	2012-10480	2012-06714
2012-13444	2012-08177	2012-05253	2012-05382	2012-05383
2012-05256	2012-06715	2012-05383	2012-06715	2012-06714
2012-07827	2012-07878	2012-05385	2012-07367	2012-06707
2012-07350	2012-04499	2012-05384	2012-13281	2012-11064
2012-10382	2012-07279	2011-5553	2012-12780	2011-2790
2010-2387	2012-11201	2012-10977	2012-11215	2012-04425
2012-09265	2012-00307	2012-04492	2012-02331	2012-10963
2012-00986	2012-4315	2012-03819		

WORK ORDERS (WO)

418123	424263	400199	396921	421701
421702	421703	417681	417698	

PROCEDURES

NUMBER	TITLE	REVISION

PROCEDURES

NUMBER	TITLE	REVISION
OP-ST-FP-0001A	Fire Protection System Inspection and Test	17
OP-ST-FP-0002	Fire Protection Water Suppression System Valve Cycling Test	33
OP-ST-FP-0011	Fire Protection System Hose Station Operability Test	8
OCAG-1	Operational Contingency Action Guideline	17
SE-ST-FP-0002	Fire Protection System Motor Driven Fire Pump Full Flow Test	21
SE-ST-FP-0003	Fire Protection System Diesel Driven Fire Pump Full Flow Test	25
FCS-65-2	Recovery Checklist Issue Closure	0
FCS-65-3	Restart Classification and Management of recovery Action Items under MC 0350 Restart Oversight	1
NP 95003 Admin C	Admin Controls for 95003 Work Scope for Station Recovery	1
PLDBD-CS-56	External Flooding	1
EPIP-OSC-7	Emergency Response Organization (ERO) Activation at the Emergency Operations Facility (EOF)	3
EPIP-EOF-1	Activation of the Emergency Operations Facility	18
EPIP-TSC-2	Catastrophic Flooding Preparations	15
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
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

PROCEDURES

NUMBER	TITLE	REVISION
OI-CW-1	Circulating Water System Normal Operation	67
	EOP/AOP Floating Steps	3

CALCULATIONS

NUMBER	TITLE	REVISION
FC 08030	Intake Structure Cell Level Control Using the Intake Structure Sluice Gates	11

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
ACA 2011-3019	Equipment Service Life Apparent Cause Analysis	1
ACA 2011-09276	Apparent Cause Analysis for Missed Vendor Manual	1
RCA 2012-03986	Organizational Effectiveness Root Cause Analysis	0
ACA 2008-05695	Apparent Cause Analysis for SI-3A-M Pump Side Motor Bearing Oil Level Found Low on Sight Glass	0
USAR 9.8	Auxiliary Systems: Raw Water System	31
RCA 2011-10135	Root Cause Analysis: Cultural Weaknesses in Problem Identification and Resolution	0
RCA 2010-2387	Root Cause Analysis: External Flooding Protection	1
LIC-11-0011	OPPD Reply to Notice of Violation EA-10-084 (Revision 1)	June 7, 2011
	Business Continuity Plan	June 11, 2011

Section 4OA5: Other Activities

DRAWINGS

NUMBER	TITLE	REVISION / DATE
11405-E-61	Reactor Auxiliary Building Tray and Conduit Layout Plan Basement FL EL 989'0 West,	Rev 51

Section 4OA5: Other Activities

DRAWINGS

NUMBER	TITLE	REVISION / DATE
11405-M-112	Containment & Auxiliary Building Miscellanious Piping Sh1,	Rev 17.
11405-M-66	Auxiliary Building RWD Vents, Drains, & Valve leak Offs EL 971'-0 and 989'-0,	Rev 19
303.130-M-001	CH-1A Oil Drain	Rev 1
70665-1 Sh1	Component Cooling Water Pump Specification,	Rev 7
A-6039 Sh 11	Safe Shutdown Target Drawing-Auxiliary Building basement Level, Room 19	Rev 0.
A-6039 Sh 20	Safe Shutdown Target Drawing –Auxiliary Building Ground Floor Level, Room 56	Rev 1
A-6039 Sh 25	Safe Shutdown Target Drawing-Auxiliary Building Ground Floor Level, Room 63	Rev 0
A-6039 SH3	Safe Shutdown Target Drawing-Auxiliary building Basement Level Room 6,	Rev 0
C 1845-833391	Installation & Assy 5 gallion 75 PSIG Suction Stabilizer.	Rev A
C-4055	Charging Pump 'A' Flushing Line Vibration Restraints	Rev 1
D-12627	Cylinder Assembly PIB-STPS,	Rev 7
D-12629	Base Outline- P18	Rev 6
D-12742	Packing Cooling System	Rev 19
D-4112,	Addition of Suction Stabilizer & Discharge Pulsation Dampener to Charging Pumps,	Rev 1
D-4228 Sh 2	CQE Piping Isometrics Seismic subsystem #CH-283-A	
FC 259	Auxiliary Building Equipment Supports,	Rev 2
FIG 8.1.1	P&ID Plant Electrical System,	Rev 142
S-53, "Auxiliary Building Intermediate Fl EL 1025'-0	Outline Sheet 1,	Rev 4

MISCELLANEOUS DOCUMENTS

TITLE	REVISION / DATE
Electric Power Research Institute document 1025286, "Seismic Walkdown Guidance,"	(ML12188A031)
IPEEE USI A46 ,"Seismic Inspections."	
List of FCS SWEL Items	9/17/12
NRC 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	(ML12053A340).
Pre- Job Brief for Fukushima NTTF 2.3 Seismic Walkdowns.	
Seismic Walkdown Checklist for AC-3B, "CCW Pump."	
Seismic Walkdown Checklist for AC-3C, "CCW Pump."	
Seismic Walkdown checklist for CH-1A, "Charging Pump"	