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

November 8, 2013

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 05000285/2013015

Dear Mr. Cortopassi:

On September 30, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. On October 25, 2013, the NRC inspectors discussed the results of this inspection with you and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

The NRC inspectors did not identify any findings or violations of more than minor significance.

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

Sincerely,

/RA/

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

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

Enclosure: NRC Inspection Report 05000285/2013015
w/Attachment: Supplemental Information

cc w/ encl: Electronic Distribution

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2013015
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	August 11 through September 30, 2013
Inspectors:	J. Kirkland, Senior Resident Inspector J. Wingebach, Resident Inspector J. Cameron, Branch Chief A. Rosebrook, Senior Project Engineer T. Brown, Senior Resident Inspector J. Laughlin, Emergency Preparedness Inspector, NSIR
Approved By:	Michael Hay, Chief Project Branch F Division of Reactor Projects

SUMMARY

IR 05000285/2013015; 08/11/2013 – 09/30/2013; Fort Calhoun Station, Integrated Resident and Regional Report; Emergency Action Level and Emergency Plan Changes

The report covered a six-week period of inspection by resident inspectors and an announced baseline inspection by one region-based inspector. No violations of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” The cross-cutting aspect is determined using Inspection Manual Chapter 0310, “Components Within the Cross-Cutting Areas.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

None

B. Licensee-Identified Violations

None

PLANT STATUS

The station began the inspection period in Mode 5 with all fuel in the reactor vessel. The reactor vessel head was set on August 23, 2013.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The NSIR headquarters staff performed an in-office review of the latest revisions of various Emergency Plan Implementing Procedures (EPIPs) and the Emergency Plan located under ADAMS accession numbers ML13165A027 and ML12363A207 as listed in the Attachment.

The licensee determined that in accordance with 10 CFR 50.54(q), the changes made in the revisions resulted in no reduction in the effectiveness of the Plan, and that the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection. The specific documents reviewed during this inspection are listed in the Attachment.

These activities constitute completion of three samples as defined in Inspection Procedure 71114.04-06.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

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

b. Findings

No findings were identified.

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

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

"Condition Report (CR) 2013-02260 identified that a summary structural analysis (FC03901) indicated that VA-15A/B (Containment Cooler/Filter Unit A/B plenum was overstressed by 100 percent and that VA-16A/B (Containment Air Cooling Unit A/B plenum) was also overstressed. At the time of discovery, FC03901 indicated that VA 15A/B required cross-bracing, which was added and the equipment was considered operable. Since VA-16A/B was overstressed, they were considered inoperable.

"During an inspection, the NRC questioned the operability determination provided in CR 2013-02260 for VA-15A/B and VA-16A/B due to the seismic criteria not being met. The station responded that since the cross-bracing had been added to VA-15A/B, they were considered operable. However, VA-16A/B did not meet the current licensing basis and they were considered inoperable. On April 6 2013, CR 2013-07674 was initiated and a reportability evaluation determined that the condition was reportable. The unit was defueled when the condition was identified.

"A causal analysis is in progress, the results of which will be published in a supplement to this licensee event report (LER)."

The LER report is closed. Revision 1 of this LER was submitted on August 16, 2013.

.2 (Opened) Licensee Event Report 05000285/2012-015-01: Electrical Equipment Impacted by High Energy Line Break Outside of Containment

"On September 16, 2011, while reviewing a draft of the Master List Reconstitution for Electrical Equipment Qualification (EEQ) (EA-FC-08-011), the Fort Calhoun Station Engineering Department identified that some of the listed components located outside of containment 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 when Fort Calhoun Station was shut down and defueled. "The causal analysis identified a number of components located in auxiliary building rooms 4, 13, 21, 22, and 81 that should have been included in the EEQ program. This omission was determined to be the result of insufficient engineering rigor by the preparer and reviewer of the EEQ Program Basis Document.

"The identified components are being qualified by additional analysis, replacement with qualified components, providing shielding or electrical isolation capabilities, or moving the component to a location where EEQ is not required."

.3 <u>(Closed) Licensee Event Report 05000285/2013-006-00:</u> Use of Teflon in LPSI and CS Pump Mechanical Seals

"On March 4, 2013, at approximately 1400 CST, it was identified that the mechanical seals used in the two low pressure safety injection pumps and the three containment spray pumps are made of a material (Teflon®) that may not maintain the designed integrity of the systems under certain accident conditions. This seal design has been installed since original plant construction. This issue was discovered by plant personnel while researching requirements for the replacement parts during scheduled outage activities.

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

The LER is closed. Revision 1 of this licensee event report was submitted on August 26, 2013.

.4 <u>(Opened) Licensee Event Report 05000285/2013-006-01:</u> Use of Teflon in LPSI and CS Pump Mechanical Seals

"On March 4, 2013, at approximately 1400 CST, while researching requirements for the replacement parts, it was identified that the mechanical seals used in the two low pressure safety injection pumps and the three containment spray pumps are made of a material (Teflon®) that may not maintain the designed integrity of the systems under certain accident conditions. This design has been installed since original plant construction. This issue was discovered while the core was off-loaded.

A causal analysis determined that Omaha Public Power District and its consulting engineering firm failed to specify a compatible material for the pump seals in the original construction specifications.

"Replacement of pump seals is scheduled to be completed prior to startup. A review of PED-GEI-10, Material Compatibility Review, will be performed to ensure the restrictions placed on the use of Teflon® are appropriate and a review of engineering change checklists will be performed to ensure the PED-GEI-10 restrictions are preserved."

.5 <u>(Closed) Licensee Event Report 05000285/2013-007-00:</u> Containment Air Cooling Units (VA-16A/B) Seismic Criteria

"CR 2013-02260 identified that a summary structural analysis (FC03901) indicated that VA-15A/B (Containment Cooler/Filter Unit A/B plenum was overstressed by 100 percent and that VA-16A/B (Containment Air Cooling Unit A/B plenum) was also overstressed. At the time of discovery, FC03901 indicated that VA-15A/B required cross-bracing, which was added and the equipment was considered operable. Since VA-16A/B was overstressed, they were considered inoperable.

"During an inspection, the NRC questioned the operability determination provided in CR 2013-02260 for VA-15A/B and VA-16A/B due to the seismic criteria not being met. The station responded that since the cross-bracing had been added to VA-15A/B, they were considered operable. However, VA-16A/B did not meet the current licensing basis and they were considered inoperable. On April 6, 2013, CR 2013-07674 was initiated and a reportability evaluation determined that the condition was reportable. The unit was defueled when the condition was identified.

"A causal analysis is in progress, the results of which will be published in a supplement to this LER."

The LER is closed. Revision 1 of this LER was submitted on August 16, 2013.

.6 (Opened) Licensee Event Report 05000285/2013-007-01: Containment Air Cooling Units (VA-16A/B) Seismic Criteria

"CR 2013-02260 identified that a summary structural analysis (FC03901) indicated that VA-15A/B (Containment Air Cooler/Filter) plenum was overstressed by 100 percent and that VA-16A/B (Containment Air Cooler) plenum would have been overstressed during a design basis seismic event. At the time of discovery, FC03901 indicated that VA-15A/B required cross-bracing, which was added resulting in the equipment being operable. Since VA-16A/B was overstressed, they were considered inoperable.

"The causal analysis determined that the design basis information was incomplete at the beginning of commercial operation. A weakness in licensing basis knowledge and a failure to internalize the importance of the design basis, resulted in the organization missing repeated opportunities to correct the initial deficiencies and additional errors were created over time. Also, the early culture established standards and expectations for the organization that resulted in behaviors demonstrating that the operation of the facility was more important than maintaining the license and design basis of the Station. This resulted in long-standing, reinforced, and institutionalized behaviors that resisted external and internal efforts to change.

"The VA-16A/B plenums will be restored to an operable condition. Additional corrective actions will be tracked by the corrective action process."

.7 (Opened) Licensee Event Report 05000285/2013-011-00: Inadequate Design for High Energy Line Break in Rooms 13 and 19 of the Auxiliary Building

"During review of the analyses for high energy line breaks two deficiencies were identified. On June 13, 2013, an unevaluated break in the steam supply to the auxiliary feedwater turbine inside Room 19 was identified. Subsequently, on June 14 2013, a deficiency was identified with verifying that the Electrical Equipment Qualification (EEQ) Program met all the criteria for establishing pipe rupture locations in Room 13.

"The Root Cause Analysis resulted in two causes. Fort Calhoun Station's responses to IE Bulletin 79-01B made inaccurate and simplifying assumptions, without supporting documentation, that compromised the validity and scope of the EEQ Program, ultimately resulting in the program being non-compliant with 10 CFR 50.49. Additionally, the EEQ Program has unique processes that are not integrated into the Engineering Change Process and impacts the sustainability of the EEQ Program.

"The required analyses will be performed, required modifications completed, and supporting documentation (including program documents) updated as required. EEQ procedures will be revised such that all EEQ engineering activities are performed under the Station's configuration change control process. Additional corrective actions will be implemented using the station's corrective action program."

.8 <u>(Opened) Licensee Event Report 05000285/2013-012-00:</u> Intake Structure Crane Seismic Qualification

On August 2, 2013, Fort Calhoun Station Engineering identified that the intake structure crane (HE-5) seismic analysis does not evaluate the crane's ability to withstand a seismic event when in use and an investigation identified that HE-5 was used when the raw water pumps were required to be operable. When discovered, Fort Calhoun Station was shutdown in MODE 5.

The crane was not in use when the condition was identified and was verified in the parked position. Compensatory actions were identified which would allow the use of the intake crane. The condition was entered into the station's corrective action program as CR 2013-15474. A new seismic analysis to address crane use will be developed.

4OA4 IMC 0350 Inspection Activities (92702)

Inspectors continued implementing IMC 0350 inspection activities, which include follow-up on the restart checklist items contained in the Confirmatory Action Letter (CAL) issued February 26, 2013 (EA-13-020, ML 13057A287). The purpose of these inspection activities 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.

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 August 11 through September 30, 2013. 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.

.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) Action Items 1.3.1.4; 1.3.1.5; 1.3.1.6; 1.3.1.15; 1.3.1.16; and 1.3.1.18

i. Inspection Scope

The Fort Calhoun Inspection Manual Chapter 0350 Restart Checklist and Basis Document dated March 7, 2013 lists actions related to the Red Finding that the NRC will verify are being adequately performed by the licensee to support plant restart. These items were also listed in the Fort Calhoun Station Flooding and Recovery Action Plan, Revision 3, dated July 9, 2012. The inspectors reviewed the following restart checklist items.

Action Item Number	Description	Status
1.3.1.4	Test all cables that terminate in 1B4A load center	Complete
1.3.1.5	Repair or replace defective cables terminating in 1B4A load center	Complete
1.3.1.6	Perform testing on the insulation of the cables that were potentially impacted by the fire located in the cable tray above 1B4A load center using EPRI technology	Complete
1.3.1.15	Provide any required Engineering Change for the non-segregated bus between 1B4A and 1B3A-4A	Complete
1.3.1.16	Repair 1B4A to 1B3A-4A non-segregated bus section	Complete

1.3.1.18	Perform testing of all circuits associated with cabling not associated with the 1B4A load center (i.e. cables located in the cable tray above the load center)	Complete

The inspectors conducted an off site review of completed engineering change packages, CRs, completed work orders, and test data to verify the above restart checklist items were adequately completed.

ii. Findings

No findings of significance were identified.

(2) Action Item 1.3.1.14

i. Inspection Scope

The purpose of Action Item 1.3.14 was to ensure the licensee cleaned equipment in the switchgear room that was coated with by-products from the fire.

The inspectors walked down the switchgear room and thoroughly inspected the room and equipment for cleanliness. Additionally, the inspectors walked down the adjacent switchgear room. The inspectors also verified the presence and condition of fire protection equipment in the rooms. The inspectors concluded that the licensee adequately cleaned the switchgear room.

This activity constitutes completion of Action Item 1.3.1.14 as listed in the Restart Checklist Basis Document.

ii. Findings

No findings were identified.

(3) VIO 2012007-01

i. Inspection Scope

The NRC issued VIO 2012007-01 for the failure of the licensee to provide adequate post-fire safe shutdown actions in the switchgear rooms. The purpose of this action item was to verify the licensee performed an adequate cause analysis and established appropriate corrective actions to address the issues.

The inspectors reviewed the documentation of the licensee's efforts. The licensee's apparent cause analysis determined the cause of the issue to be that the verification and validation process AOP-6, "Fire Emergency," did not have enough rigor because the guidance could introduce human performance errors. The licensee's corrective actions include performance of a complete walkdown of AOP-06 to ensure it is able to be performed as written; revision of the verification and validation walkdown procedure for AOP-06 to require a walkdown for all procedure changes; and discussion of AOP-06 changes with the Fire Protection Focus Team. The inspectors concluded the cause analysis and corrective actions appear adequate to minimize recurrence of the issue.

This activity constitutes closure of VIO 2012007-01 as listed in the Restart Checklist Basis Document.

ii. Findings

No findings were identified.

.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 appropriate condition to support safe restart and continued safe plant operation.

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

Item 2.a is the NRC's independent evaluation of Fort Cahloun 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.

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

(1) CAL Action Item 2.3.1.8

i. Inspection Scope

The purpose of Action Item 2.3.1.8 was to start the motor for circulating water pump CW-1B and take vibration data. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

Action item 2.3.1.4 was completed to determine whether circulating water pump motors were to be refurbished or if they were adequate for use. It was determined that the A and C Circulating Water Pumps were to be refurbished. This action item was inspected and addressed in inspection report 05000285/2012-005 (ML13164A359). The B Circulating Water Pump was determined to not be refurbished, thus the motor was static tested. The motor static test was inspected in conjunction with action item 2.3.1.7, and was documented in inspection report 05000285/2012-014 (ML13266A142).

The licensee started the B Circulating Water Pump motor on June 22, 2013. The inspectors witnessed the starting of the pump, and independently reviewed the vibration data results and verified that the results were within specification as required by the surveillance test.

This activity constitutes completion of Action Item 2.3.1.8 as described in CAL EA-13-020.

ii. Findings

No findings were identified.

(2) CAL Action Item 3.1.1.1

i. Inspection Scope

The purpose of Action Item 3.1.1.1 was to document review of all Engineering Programs. This item was required to be completed prior to exceeding 210 degrees Fahrenheit in the Reactor Coolant System.

The licensee evaluated the 41 Engineering programs against eight questions:

1. Have any of the program components come in contact with flood water?
2. Does river water normally come in contact with the components during normal operation?
3. Could flood water that came in contact with the program component have any additional negative effect other than normal operation?
4. Did the condition of the river water flooding change river water properties to a point that is different than expected and could result in a negative effect on a program component?
5. Could the extended period of time of river flooding conditions result in a cumulative negative effect on the component?
6. Could the added elevation of the river have a negative effect on the program components?
7. Are there any flooding CRs that would not be covered by the system review?
8. Are there any flooding temporary modifications that would not be covered by the system review?

Eight programs screened as having been potentially impacted by the flood:

1. Service Water Reliability Program (answered yes to questions 1-5);
2. Motor Maintenance and Monitoring Program (answered yes to questions 1, 3, and 5);
3. Dry Fuel Storage Program (answered yes to questions 1, 3, 5, and 6);
4. Buried Piping and Components Program (answered yes to questions 1-5);
5. Cables and Connections Program (answered yes to questions 1, 3, 5, and 6);
6. Groundwater Protection Program (answered yes to questions 1 and 3);
7. Structures Monitoring Program (answered yes to questions 1-3, 5, and 6);
8. Fire Protection Program (answered yes to questions 1-5).

The overall flood recovery action plan was used to address the issues that were identified in the review of the programs.

The inspectors reviewed the results of the screening performed by the licensee, and concluded that the actions taken by the licensee to ensure that the components affected in the affected programs have been adequately addressed.

This activity constitutes completion of Action Item 3.1.1.1 as described in CAL EA-13-020.

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.

.a Corrective Action Program

(1) Item 3.a.13: Raw Water Pump A-10C High Vibrations

i. Inspection Scope

The inspectors evaluated the adequacy of the licensee's assessment of the identification of high vibrations on raw water pump A-10C in August 2012. The issue was documented in CR 2012-12067, "AC-10C Raw Water Pump available but not Operable do to high motor vibrations," and was subject to an apparent cause analysis (ACA). The ACA was completed on February 19, 2013. (CL Item 3.a.13)

ii. Observations and Findings

Determine that the problem was evaluated using a systematic methodology to identify the apparent causes.

The inspectors determined that the licensee evaluated the problem using several systematic methodologies to identify the apparent causes. The licensee used the following systematic methods to complete the ACA report: (1) Fault Tree Analysis, (2) Support Refute Analysis, and (3) Human Performance Evaluation System were selected based on the equipment and human performance issues identified.

Determine that the apparent cause evaluation was conducted to a level of detail commensurate with the significance of the problem.

The inspectors determined that the licensee conducted the ACA to a level of detail commensurate with the significance of the problem. The licensee identified one apparent cause for the condition that existed, which was low engineering rigor resulted in substitute replacement item (SRI) for an engineering change

(EC 34242) for the “C” raw water pump omitting a specification to install a stainless steel lock washer. Instead, a carbon steel lock washer was installed. Station drawings indicate that the suction bell, series case, top case, and flange connection fasteners: stud and hex nut for the raw water pump are all fabricated from stainless steel material. The material of the lock washer is not specified on the drawing file, but research of the last pump rebuild (prior to the pump being put into service in 2008) indicated that the washer was fabricated from carbon steel. When the suction bell, series case and top case flanges are bolted together with the carbon steel lock washer and placed in a wetted environment, a galvanic cell is created. The carbon steel lock washer is the more active metal (anode) and was subjected to galvanic corrosion. The licensee determined that the accelerated rate of galvanic corrosion was due to the large ratio of stainless steel to carbon steel. As the carbon steel washer corroded and was no longer able to perform its function, flange compression was lost, which resulted in the loose nuts, flange separation and increased movement of the suction bell and series case. The loose parts, as a result of the failed lock washer, would account for the large step change in vibration that was observed during testing.

Determine that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience.

The inspectors determined that the ACA included a consideration of prior occurrences of the problem and knowledge of prior operating experience. The licensee identified the potential for galvanic corrosion in its search of internal operating experience due to dissimilar metals on the raw water pumps in 2004. At that time, during a rebuild of the “B” raw water pump, the licensee identified several fasteners that were severely corroded. The corrosion was attributed to a galvanic reaction due to dissimilar metals in a wetted environment. As a result, the licensee implemented Engineering Change 34242, "Install Stainless Steel Fasteners on RW Pump Cases,” on June 15, 2004. Although this EC was put into practice for the specific purpose of eliminating galvanic corrosion, the workers who rebuilt the “C” pump failed to install a stainless steel lock washer, because the material properties of the washer were not specified in the work order. The licensee queried industry operating experience to search for issues related to service water pumps, bolting, and fasteners. The review focused on industry experience from about October 2007 to October 2012. There is no industry operating experience found that was similar (using dissimilar material fasteners) to that experienced by the licensee.

Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem.

The inspectors determined that the licensee’s ACA addressed the extent of condition and the extent of cause of the problem. The inspectors determined that the scope of the ACA focused on other pumps that used raw water and may have dissimilar metals in a wetted environment. The licensee’s extent of condition evaluated fire water and circulating water pumps and determined that the material specifications for those pumps did not introduce a dissimilar metal condition. The licensee determined that the extent of condition was limited to the other three raw water pumps.

Determine that the root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in IMC 0310.

Based on the licensee's procedures (FCSG-24-4, "), ACA do not include an evaluation of the safety culture components. However, the ACA did address the human performance component of the issue. The results of the ACA determined that the issue was attributable to an engineering human performance error. This engineering error was consequential because the planner used the substitute replacement item equivalency evaluation (which included the drawing for the raw water pump) for parts and incorporated an existing work order to replace the fasteners with stainless steel, but did not include lock washers. The comments page of the work order did not list the lock washer as a replaced part. The licensee assumed that the original lock washer was reused.

Determine that appropriate corrective actions are specified for each root and contributing cause.

The inspectors determined that the licensee specified appropriate corrective actions for the apparent cause. The corrective actions included providing an updated Engineering Change to specify a stainless steel lock washer to be used when rebuilding the raw water pumps, a historical review of past Engineering Changes for the raw water pumps to potentially identify any other dissimilar metal conditions, a vibration monitoring program for the remaining three raw water pumps pending inspection and replacement of the lock washers, and a schedule for inspection and replacement of the lock washer with one of stainless steel on those other pumps.

Determine that a schedule has been established for implementing and completing the corrective actions.

The inspectors determined that the licensee established a schedule for implementing and completing the corrective actions. All but one of the corrective actions were implemented by May 2013. The remaining corrective action, for the inspection and replacement of the lock washers on the "A," "B," and "D" raw water pumps was scheduled to be completed by November 28, 2013.

Determine that quantitative or qualitative measures of success have been developed for determining the effectiveness of the corrective actions to prevent recurrence.

Quantitative or qualitative measures were not developed, or required, to support an effectiveness review of the corrective actions proposed by this ACA.

iii. Assessment Results

The inspectors concluded that the licensee understood the apparent cause of this issue associated with high vibrations on the “C” raw water pump, and had instituted appropriate corrective actions to prevent recurrence. Therefore, restart checklist item 3.a.13 will be closed.

(2) Item 3.a.14 and 3.a.15: Reactor Cavity Leakage and Effects on Equipment

(i) Inspection Scope

The team evaluated the adequacy of the licensee’s assessment of the reactor cavity leakage and the potential impact on other equipment between the leak location and the ground floor of containment (994 feet), the thoroughness of their extent of condition and causal analysis, and the adequacy of identified corrective actions to stop the leakage (CL Items 3.a.14; 3.a.15). The apparent cause report addressed licensee condition report CR 2012-0116 “Reactor cavity liner leakage has resulted in the accumulation of boric acid in several locations in containment.”

(ii) Observations and Findings

Determine that the problem was evaluated using a systematic methodology to identify the root and contributing causes.

The inspector determined that the licensee evaluated the problem using appropriate methodologies to identify the source of the reactor cavity leakage. The licensee’s evaluation was limited to an apparent cause, rather than a full root cause analysis based on the significance coding of the condition in its corrective action program (Level 2 – an event that results in moderate impact). The inspectors determined that the significance coding and level of investigation were appropriate for the condition.

Determine that the apparent cause evaluation was conducted to a level of detail commensurate with the significance of the problem.

The team determined that the licensee conducted the apparent cause evaluation to a level of detail commensurate with the significance of the problem. The licensee identified three apparent causes for the condition that existed, which were the refueling cavity reactor flange seal, from one or more of the six hot and cold leg inspection sandbox seals, or through the concrete that makes up the refueling cavity. Based on visual inspections, the licensee determined that the most probable source of the leakage was from the sand box covers/gaskets.

Determine that the apparent cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience.

The team determined that the apparent cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience. The licensee identified occurrences and operating experience of the problem as a part of the review of internal and external operating experience. The review of internal operating experience identified instances of reactor cavity leakage as far back as 2005, when leakage took a step change increase from approximately 10 quarts per day to 192 quarts per day. The lower levels of leakage had been routinely identified as far back as February 2000 as part of a preventative maintenance activity. The licensee identified these instances as missed opportunities.

Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem.

The team determined that the licensee's apparent cause evaluation did not fully address the extent of condition of the problem. The extent of cause evaluation was considered to be acceptable. The team determined that the scope of the extent of condition evaluation focused mainly on Electric Power Research Institute Document No. 1022880, "Welding and Repair Technology Center: Boric Acid Attack of Concrete and Reinforcing Steel in PWR Fuel-Handling Buildings." However, based on NRC inspection, and interviews with the licensee's recovery team members, the EPRI document addresses potential concrete degradation in a static condition (concrete soaking in a boric acid bath, without flow) and the impact of boric acid on concrete with non-reactive aggregate. The concrete in the structures around the reactor cavity at Fort Calhoun Station contains limestone aggregate. The EPRI document specifically describes such aggregate as "reactive" in nature. Therefore, the conclusions drawn from the EPRI document are not analogous to the conditions at Fort Calhoun Station and the potential consequences of reactor cavity leakage. In particular, the apparent cause report did not address the effects of flow on the debonding of the cement paste of the concrete, did not confirm the pH of the leakage from the concrete at the lower levels of containment to support the conclusion that the degradation of concrete reinforcement could be expected to be minimal, failed to address the potential for alkali silica reaction, and failed to address the potential effects on other equipment.

In response to these questions posed by the inspector, the licensee planned to supplement or revise the apparent cause report. However, as of the end of the report period the applicable supplement/revision had not been completed.

Determine that appropriate corrective actions are specified for each root and contributing cause.

The team determined that the licensee specified appropriate corrective actions for the apparent cause. The initial corrective action included a work order to replace the gasket on the sand box and nuclear detector wells. However, following completion of the work order, reactor cavity leakage increased from 700 gallons per day to approximately 1,300 gallons per day. The revised corrective actions include welding the sand box covers to preclude further leakage.

Determine that a schedule has been established for implementing and completing the corrective actions.

The team determined that the licensee established a schedule for implementing and completing the corrective actions. Due to resource constraints in the current outage, welding of the sand box covers will not occur until the next refueling outage, and prior to flooding the refueling cavity. Since there currently was no water inventory in the refueling cavity there is no ongoing leakage. The team questioned the timing of the future welding operations, considering radiation levels are currently lower than they would be immediately following shutdown at the time of the next refueling outage. The licensee considered this, but ultimately determined that insufficient resources were available to complete welding operations during the current extended outage.

(iii) Assessment Results

The team concluded that the licensee understood the apparent causes of this issue associated with reactor cavity leakage. However, the licensee failed to adequately address the extent of condition for the reactor cavity leakage and did not adequately support its conclusion that the leakage would have little or no impact on the concrete structures around the refueling cavity. Based on the inspector's questions, the licensee planned to either supplement or revise its apparent cause report. At the end of the inspection period neither the supplement or revision had been completed. Therefore, restart checklist items 3.a.14 and 3.a.15 will remain open. The inspector reviewed operating experience related to long-term leakage at other reactor sites, in particular Connecticut Yankee, where boric acid leakage through concrete structures occurred throughout that plant's operating life. During decommissioning, analysis of the affected concrete determined that there was no significant degradation of the concrete from the boric acid leakage. As a result of that review, the inspector determined that the likelihood of significant degradation of concrete structures at Fort Calhoun Station from reactor cavity leakage were low, and for that reason, the resolution of the impact of reactor cavity leakage on concrete and other equipment was not considered a heatup or startup restraint. Therefore, resolution of those issues will be completed following reactor startup.

.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.

(2) Action Item 4.6.1.2

i. Inspection Scope

The purpose of Action Item 4.6.1.2 was to address non-conservative 161kV minimum voltage to support operation of a main feedwater pump in a Safety Injection Actuation signal (SIAS)-only scenario.

The inspectors reviewed the documentation of the licensee's efforts. The licensee issued a new engineering analysis, EA11-048, to calculate the minimum 161kV voltage with one main feedwater pump in operation that would not result in an Offsite Power Low Signal (OPLS) during a SIAS-only event. The licensee revised plant operating procedures and alarm setpoints to reflect the new voltage requirements to ensure offsite voltage remains operable and direct operator actions if voltage drops below the new minimum setpoint.

This activity constitutes completion of Action Item 4.6.1.2 as listed in the Restart Checklist Basis Document.

ii. Findings

No findings were identified.

.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.

(1) Vendor Modification Control

i. Inspection Scope

The inspectors evaluated the adequacy of the licensee's assessment of the Vendor Modification Control program, the thoroughness of their extent of condition and extent of cause analysis, and the adequacy of identified corrective actions to ensure proper control of vendor modifications to the facility.
(CL Items 3.c.1.1; 3.c.1.2; 3.c.1.3)

ii. Observations and Findings

Determine that the problem was evaluated using a systematic methodology to identify the root and contributing causes.

The inspectors determined that the licensee relied heavily on the root cause evaluations conducted for CR 2011-5414, "Breaker Cubicle 1B4A Fire (Red Finding)," CR 2011-6621, "1B3A Main Breaker Trip During Switchgear Fault on 1B4A," CR 2012-07279, "Vendor Modification Collective Significance," and

CR 2013-05570, "Design and Licensing Bases Configuration Control." Each of those evaluations were conducted using systematic methodologies to identify root and contributing causes for the issues related to the applicable Condition Report. In turn, the licensee associated those root and contributing causes to the weaknesses identified related to Vendor Modification Control. The inspectors determined that this was an acceptable approach.

Determine that the root cause evaluation was conducted to a level of detail commensurate with the significance of the problem.

As described previously, the licensee's evaluation of the Vendor Modification Control program relied upon evaluations conducted for other related activities that pointed to the overall weakness within this program. The inspectors determined that the licensee conducted the root cause evaluations to a level of detail commensurate with the significance of the associated problems. The licensee identified four root causes for the Vendor Modification Control program that were taken from two of the other associated root cause evaluations: from CR 2011-5414 (RC-1 – The design process failed to identify critical parameters and interfaces such as the silver plating contact area on the switchgear cubicle stabs; RC-2 – Design Change Package (DCP) preparation procedures do not provide guidance to evaluate design features of new components in regard to the possibility that they may adversely affect required performance characteristics if not properly configured.) and from CR 2013-05570 (RC-1 – OPPD Design and Licensing Bases information was incomplete at the beginning of commercial operation; RC-2 – The early culture established standards and expectations for the organization that resulted in behaviors demonstrating that the operation of the facility was more important than maintaining the license and design basis of the station).

Determine that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience.

The inspectors determined that the root cause evaluations for the issues associated with the Vendor Modification Control program included a consideration of prior occurrences of the problem and knowledge of prior operating experience. The licensee identified occurrences and operating experience of the problem as a part of the root cause evaluations for the other associated CRs, most notably CR 2013-05570, "Design and Licensing Bases Configuration Control." As part of the operating experience review for that root cause evaluation, the licensee identified sixteen previous examples of internal operating experience contained within correspondence between the licensee and the NRC regarding missed vendor audits, NRC-identified violations regarding design control, weaknesses in the interface between plant QA and licensing departments, and other similar issues. Each of these occurrences was identified as a missed opportunity by the licensee. As part of its review of external operating experience, the licensee identified a number of vendor-related design issues and deficiencies. Overall, the licnesee's operating experience review determined that an over-reliance on vendor knowledge and skill was an industry issue. The NRC has also highlighted this issue to the industry through the issuance of two related Generic Letters that address Licensee/Vendor interface.

Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem.

The inspectors determined that the licensee's root cause evaluation did not fully address the extent of condition of the problem. The original Discovery Phase scope evaluated 49 modification packages completed between 2007 and 2012. The evaluation did not identify any deficiencies similar to those addressed by the four root cause evaluations performed for associated issues. Nevertheless, the licensee expanded the extent of condition review to include all modification packages completed between April 2007 and April 2013 (57 in total); the results of a contractor review in 2010 of 100 design packages completed from October 2009 through December 2010; 13 modifications and field changes that had been designed, issued, and installed during the 2011 refueling outage and had not yet been tested; an effectiveness review of 19 modification completed between August 2012 and April 2013; and 1304 modifications reviewed by Stone and Webster in 1990 as part of a SEP commitment to the NRC and included those modifications completed from 1974 through 1988 in an effort to assess the technical adequacy of the OPPD 10 CFR 50.59 process.

The expansion review identified a number of issues with the design and modification packages that were binned as Administrative (such as typographical and grammatical errors), Technical (such as missing, not clearly defined, or not fully analyzed critical characteristics), and Programmatic (such as inadequately maintained design basis). None of the issues identified called into question the technical adequacy of the completed engineering changes, much less rose to the level of significance as the Breaker Cubicle 1B4A Fire. However, the contractor review conducted in 1990 identified 16 unreviewed safety questions (USQ) (issues that would indicate the need for NRC approval prior to implementing the proposed change). Those 16 USQs were all related to pipe support/pipe stresses. The contractor's report indicated that OPPD was resolving the 16 USQs in accordance with plant procedures at the time of the review. A licensee search of the closure documentation could not determine how three of them were resolved. The remainder were closed satisfactorily. The licensee entered those three issues in the Corrective Action Program and were being tracked for resolution. The licensee eventually located the closure information for the remaining three USQs.

In addition to the 16 USQs, the 1990 contractor report included mention of 197 modification packages for which insufficient data was available to make a USQ/Technical Specification determination. These included code reconciliation issues, seismic qualifications, seismic II over I, IE Bulletin 79-14/79-02 (pipe stresses and supports), Technical Specification components/documentation conflicts, passive safety-related (non-TS) components missing documentation, an industry/NRC generic issue, and one package awaiting paperwork close-out. None of these issues are addressed in the licensee's evaluation of the Vendor Modification Control program. A search of CAP documents identified CR 2013-07316, "Potential Unreviewed Safety Questions" initiated on April 2, 2013. For the CR, the licensee selected four packages for review. No outstanding USQs were identified. However, the licensee's evaluation of the extent of condition failed to address these 197 modification packages, and there was no immediate plan to address the 193 packages that were not selected for review under CR 2013-07316. The licensee opened CR 2013-18229 to address the inspector's question. The licensee subsequently screened the remaining modification packages and determined that a majority of the open items associated with the 197 indeterminate modifications identified in the contractor's 1990 report have been completed and documented as such within the Design Basis Open Item Closure Project Action Plan dated August 23, 1996, with no USQs identified.

The licensee generated an action item for CR 2013-18229 to review and verify that all open items from the 1990 contractor report have been closed prior to plant restart. Any items not closed will be entered in the Corrective Action Program with actions to close prior to plant restart. Furthermore, the licensee will review for technical rigor and adequacy the closure documentation of the 197 indeterminate modifications. This action will be completed as part of the Design Basis Reconstitution Project, which will occur after restart. In addition, the inspector selected four modification packages that were flagged as indeterminate in the 1990 contractor's report for review. The inspector's review did not identify any issues of concern.

Determine that the root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in IMC 0310.

The inspectors determined that the root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in IMC 0310 for the four root causes for the issues associated with this Restart Issue. The licensee reviewed each safety culture component and determined whether the condition was applicable so that they could link the component to a root or contributing cause.

Determine that appropriate corrective actions are specified for each root and contributing cause.

The inspectors determined that the licensee specified appropriate corrective actions for each root and contributing cause. The proposed corrective actions include both interim and permanent actions as the site transitions to Exelon fleet model for Engineering Changes. The corrective actions include changes to procedures, processes and training, and user desktop guides for engineers involved in preparing and reviewing Engineering Changes. The interim actions appear to adequately address the gaps between the current station program and the proposed fleet model used by Exelon. At the time of the inspection, the licensee was in the process of implementing the interim corrective actions.

Determine that a schedule has been established for implementing and completing the corrective actions.

The inspectors determined that the licensee established a schedule for implementing and completing the corrective actions. The schedule included both the interim and permanent corrective actions to address the gaps between the station's current program and fleet model used by Exelon.

Determine that quantitative or qualitative measures of success have been developed for determining the effectiveness of the corrective actions to prevent recurrence

The inspectors determined that the licensee developed quantitative and qualitative measures of success for determining the effectiveness of the corrective actions to prevent recurrence. These effectiveness reviews consisted of developing performance indicators to track and analyze trends of performance gaps in other areas, such as the licensee's 10 CFR 50.59 process.

iii. Assessment Results

The inspectors concluded, for the most part, that the licensee thoroughly assessed and developed adequate corrective actions to address the root and contributing causes of the fundamental performance deficiency associated with the Vendor Modification Control program. Therefore, Restart Checklist Items 3.c.1.1 and 3.c.1.3 will be closed.

The inspectors concluded that the licensee initially failed to fully evaluate the extent of condition, in that the assessment did not address 197 modification packages identified by a contractor during an independent assessment in 1990 for which insufficient information was available at that time to determine whether an Unreviewed Safety Question existed. However, during the inspection, the licensee was able to demonstrate that adequate closure information was available to determine that an Unreviewed Safety Question did not exist as a result of those modifications. An independent review by the inspectors of selected modification packages did not identify any issues of concern. Therefore, Restart Checklist Item 3.c.1.2 will be closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 25, 2013, the inspectors presented the inspection results to Mr. L. Cortopassi, Senior Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

4OA7 Licensee-Identified Violations

None

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Bakalar, Manager, Security
J. Bousum, Manager, Emergency Planning and Administration
C. Cameron, Supervisor Regulatory Compliance
L. Cortopassi, Site Vice President
K. Ihnen, Manager, Site Nuclear Oversight
T. Leeper, Manager, Human Resource Services
T. Lindsey, Director, Training
E. Matzke, Senior Licensing Engineer, Regulatory Assurance
B. Obermeyer, Manager, Corrective Action Program
T. Orth, Director, Site Work Management
E. Plautz, Supervisor, Emergency Planning
R. Short, Assistant Director, Engineering
T. Simpkin, Manager, Site Regulatory Assurance
S. Swanson, Manager, Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285-2012-015-01	LER	Electrical Equipment Impacted by High Energy Line Break Outside of Containment (Section 4OA3)
05000285-2013-006-01	LER	Use of Teflon in LPSI and CS Pump Mechanical Seals (Section 4OA3)
05000285-2013-007-01	LER	Containment Air Cooling Units (VA-16A/B) Seismic Criteria (Section 4OA3)
05000285-2013-011-00	LER	Inadequate Design for High Energy Line Break in Rooms 13 and 19 of the Auxiliary Building (Section 4OA3)
05000285-2013-012-00	LER	Intake Structure Crane Seismic Qualification (Section 4OA3)

Closed

05000285-2012-015-00	LER	Electrical Equipment Impacted by High Energy Line Break Outside of Containment (Section 4OA3)
05000285-2013-006-00	LER	Use of Teflon in LPSI and CS Pump Mechanical Seals (Section 4OA3)
05000285-2013-007-00	LER	Containment Air Cooling Units (VA-16A/B) Seismic Criteria (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1EP4: Emergency Action Level and Emergency Plan Changes

PROCEDURES

NUMBER	TITLE	REVISION
EPIP-EOF-3	Offsite Monitoring	27
EPIP-OSC-1	Emergency Classification	48
	Evacuation Time Estimate Study Update	

Section 4OA2: Problem Identification and Resolution (71152)

PROCEDURES

NUMBER	TITLE	REVISION
FCSG-24-1	Condition Report Initiation	5
FCSG-24-3	Condition Report Screening	7
FCSG-24-4	Condition Report and Cause Evaluation	7
FCSG-24-6	Corrective Action Implementation and Condition Report Closure	10
SO-R-2	Condition Reporting and Corrective Action	53b

Section 4OA4: IMC 0350 Inspection Activities (92702)

CONDITION REPORTS (CR)

2011-06725	2012-05766	2013-01917	2013-01918	2012-05850
2013-02041	2009-02306	2010-05140	2013-03024	2012-05766
2012-05854	2012-05855	2011-6621	2011-5414	2011-8957
2013-16104	2011-8958	2013-18229	2011-5414	2011-6621
2012-07279	2013-05570	2013-07316	2012-08125	2009-01851
2011-03025	2012-03986	2012-08134	2012-08136	2012-08137

WORK ORDERS (WO)

400051	415420	419148	419149	419150
419151	419163	419164	419171	419173
419174	420350	420768	421702	444700

489275

PROCEDURES

NUMBER	TITLE	REVISION
AOP-31	161kV Grid Malfunctions	13
EM-PM-EX-1001	4160 Volt Motor Inspection	23, 26
FCSG-24-4	Condition Report and Cause Evaluation	8
FCSG-24-4	Condition Report and Cause Evaluation	7
OI-EG-3	EMS Post-FCS-Trip 161kV Voltage Prediction and Switchyard Status	13
PED-GEI-29	Preparation of Facility Changes	58
PED-GEI-3	Preparation of Modifications	92
PED-GEI-35	Preparation of Minor Configuration Changes	72
PED-GEI-44	Processing Configuration Changes	19
PED-GEI-52	Preparation of Field Design Change Requests (FDCR)	15
PED-GEI-56	Configuration Change Closeout	30
PED-QP-11	Independent Design Verification (IDV) and Independent Review of Configuration Changes	12
PED-QP-13	Design Basis Documents	8
PED-QP-2	Configuration Change Control	61
PED-QP-5	Engineering Analysis	45
PED-QP-6	Procurement / Specifications	21
SO-C-2	Quality Assurance Records	99
SO-G-21	Station Modification Control	96
SO-G-74	Fort Calhoun Station EOP/AOP Generation Program	17
SO-O-1	Conduct of Operations	101
SO-O-25	Temporary Modification Control	84

DRAWINGS

NUMBER	TITLE	REVISION
SK-EC-53517-1	Auxiliary Building EL 1011’-0” Cable Tray Repair Detail	B

ENGINEERING CHANGE (EC)

NUMBER	TITLE	REVISION
53257	480V Bus 1B4A Repair / Replacement	0
53517	Repair of 1B4A Fire Damaged Cables	1
34242	Install Stainless Steel Fasteners on RW Pump Cases	0
57990	Upgrade RW Pumps AC-10A/B/C/D Fasteners to Stainless Steel	2

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	Letter From A. Mantey To Merl Core Subj: Cable Inspection and Intentor Testing at Ft. Calhoun Station	8/1/2011
	Tyco Electronics Raychem WCSF Datasheet	
	WCSF-N Installation Instructions	
13-4060-01	Omaha Public Power District Fort Calhoun Nuclear Generating Station Unreviewed Safety Question Report, June 11, 2013	0
17321.15-L(D)-1	The Modification Safety Evaluation Review, December 1990	0
EA-11-048	Post-Trip Degraded Voltage for Main Feedwater Pump Operation	0
EA-92-081	OPLS Setpoints for Operation of a Main Feedwater Pump	1
EA-FC-90-057	Updated Degraded Voltage Calc 4160V/480V	10
EA-FC-90-076	Cable Trap Loading Calc/Justification	12
EA-FC-91-36	FW/Condensate Pump Operation (OPLS)	0
IEEE 308	Standard for Class 1E Power Systems For Nuclear Power Generating Stations	1974
IEEE 383	Standard for Type Test of Class 1E Electrical Cables, Field Splices, and Connections for Nuclear Power Generating Stations	1974
NED-11-0097-DEN	Letter From S.Miller Subj: Minor Revision to EA-FC-90-076 per EC 53517	8/17/2011
Recovery Issue 3.c.1	Vendor Modification Control, September 10, 2013: Resolution Narrative, Tab B; Causal Analysis Summary,	

MISCELLANEOUS DOCUMENTS

	Tab H; Extent of Condition Discussion, Tab I.	
SDBD-EE-201	AC Distribution	24
USAR Appendix G	Responses to 70 Criteria	21
USAR Section 6	Engineered Safeguards	
USAR Section 7	Instrumentation and Control	
USAR Section 8	Electrical Systems	