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

April 3, 2014

EA-13-201

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

SUBJECT: FORT CALHOUN – MANUAL CHAPTER 0350 TEAM INSPECTION REPORT NO. 05000285/2013013 AND NOTICE OF VIOLATION

Dear Mr. Cortopassi:

On February 18, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed a team inspection at the Fort Calhoun Station (FCS). The purpose of this inspection was to evaluate the readiness of plant hardware, plant staff, plant processes, and management programs that supported safe restart and continued operation of the FCS. The team focused on those issues described in the Restart Checklist, enclosed in the Confirmatory Action Letter issued to the FCS on June 11, 2012 (ML12163A287), and updated on February 26, 2013 (ML13057A287), which were ready for NRC inspection. The enclosed report documents the inspection results which were discussed on February 18, 2014, with you and other members of your staff.

During this inspection, the NRC staff 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 team reviewed selected procedures and records, observed activities, and interviewed personnel.

Twenty one findings of very low safety significance (Green) are documented in this report. All of these findings involved violations of NRC requirements. Three of these violations were determined to be Severity Level IV under the traditional enforcement process. One of the SLIV violations is being cited in the enclosed Notice of Violation (Notice) as discussed below.

The NRC determined that a Severity Level IV violation of NRC requirements occurred. The circumstances of the violation involved incomplete and inaccurate information submitted by FCS in a response to a Request for Additional Information (RAI) concerning the exemption request from the requirements of 10 CFR Part 50, Appendix R, Section III.G.1.b for Fire Area 31 at the Fort Calhoun Station. The details of the violation are described in the enclosed report. The violation was evaluated in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC's web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. In accordance with Section 6.9.c.1 of the Enforcement Policy, this violation would normally be assessed as Severity

Level III. However, in accordance with the Enforcement Policy, and considering the very low safety significance (Green) of the associated finding, the NRC concluded this violation is more appropriately assessed as Severity Level IV with a response required.

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's review of your response to the notice will also determine whether further enforcement action is necessary to ensure your compliance with regulatory requirements.

If you contest these violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the 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 the Fort Calhoun Station.

If you disagree with a cross-cutting aspects 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 the FCS.

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

Sincerely,

/RA/

Michael Hay, Chief
Project Branch F
Division of Reactor Projects

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

Enclosure:
1. Notice of Violation
2. NRC Inspection Report 05000285/2013017
   w/Attachments:
   Attachment 1: Supplemental Information

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ROP reports

File located: R:_REACTORS_FCS\2014 ADAMS: ML14094A052

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

SRI:DRP/C	SRI:DRS/EB2	RI:DRS/EB2	BC:DRS/EB2	ORA/ACES
JJosey	SGraves	JWatkins	JDixon	RBrowder
/RA/	/RA/	/RA/	/RA/	/RA/
4/3/14	3/26/14	4/2/14	4/2/14	4/1/14
BC:DRP/F				
MHay				
4/3/14				
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NOTICE OF VIOLATION

Omaha Public Power District (OPPD)
Fort Calhoun Station

Docket No. 50-285
License No. DPR-40
EA-2013-201

During a U.S. Nuclear Regulatory Commission (NRC) inspection conducted from July 8 through December 16, 2013, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR 50.9(a), "Completeness and Accuracy of Information," requires in part that, "information provided to the Commission by a licensee shall be complete and accurate in all material respects."
>
> Contrary to the above, on October 13, 2008, the licensee provided to the Commission documentation which contained information that was not complete and accurate in all material respects. Specifically, the licensee submitted a letter dated October 13, 2008, which stated that the pyrocrete enclosure remained in place to protect the cables associated with AC-10A and AC-10B from a fire in the intake structure. When in fact, the motor lead cables associated with raw water pump AC-10A were not protected by the pyrocrete enclosure. In a letter, dated February 6, 2009, the NRC granted an exemption from the specific requirements of Section III.G.1.b of 10 CFR Part 50, Appendix R, for the Fort Calhoun Station based in part, upon the NRC's review and evaluation of information provided by the licensee in its letter dated October 13, 2008. Therefore, this information was considered material to the NRC.

This is a Severity Level IV Violation (Section 6.9).

Pursuant to the provisions of 10 CFR 2.201, Omaha Public Power District (OPPD) 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 at the Fort Calhoun facility, 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-13-201" and should include for the 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 Agencywide Documents Access and Management 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 or proprietary 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).

Dated this 3rd day of April 2014.

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2013013
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	July 8, 2013 through February 18, 2014
Inspectors:	H. Barrett, Senior Fire Protection Engineer, Headquarters R. Deese, Senior Project Engineer, Region IV G. George, Senior Reactor Inspector, Region IV J. Hanna, Senior Reactor Analyst, Region II R. Haskell, Reactor System Engineer, Headquarters C. Henderson, Resident Inspector, Region IV J. Jacobson, Senior Reactor Operations Engineer, Headquarters J. Josey, Senior Resident Inspector, Region IV S. Laur, Senior Reliability and Risk Analyst, Headquarters T. Lightly, Project Engineer, Region II D. Loveless, Senior Reactor Analyst, Region IV S. Makor, Reactor Inspector, Region IV J. Polickoski, Project Manager, Headquarters F. Ramirez, Resident Inspector, Region III J. Robles, Reactor System Engineer, Headquarters C. Sanders, Allegations Specialist, Headquarters A. Scarbeary, Resident Inspector, Region III C. Smith, Project Engineer, Region IV R. Telson, Reactor Operations Engineer, Headquarters J. Watkins, Reactor Inspector, Region IV J. Wingebach, Resident Inspector, Region IV
Accompanying Personnel	C. Baron, Mechanical Contractor, Beckman and Associates N. Patel, Electrical Contractor, Beckman and Associates
Approved By:	Michael Hay, Chief Project Branch F Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2013013; 07/08/2013 – 2/18/2014; Fort Calhoun Station, Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs or One Red Input.

The report covered a seven month period of inspection by an Inspection Manual Chapter 0350 inspection team. Eighteen Green non-cited violations were identified. Additionally, one cited and two non-cited, Severity Level IV violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The cross-cutting aspect is determined using Inspection Manual Chapter 0310, "Components Within the Cross Cutting Areas." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a non-cited violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Actions," for the licensee's failure to promptly identify and correct a condition adverse to quality. Specifically, the licensee failed to fully implement a corrective action from a previous breaker issue, which was to perform current injection testing for the 480 Vac 1B4A bus breakers without the full function test kit. Testing with the full function test kit would not identify if zone select interface jumpers were incorrectly installed. The licensee performed current injection testing without the full functional test kit on the 480 Vac load center main breaker 1B4A and the bus tie breaker BT-1B4A. The licensee addressed this deficiency by performing the appropriate testing on the two breakers. The licensee entered this deficiency into their corrective action program for resolution as Condition Report (CR) 2013-13262.

  The licensee's failure to promptly identify and correct a condition adverse to quality is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the associated objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 4, "PWR Refueling Operation: RCS level >23' or PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, and determined that the finding is of very low safety significance (Green) because the finding did not require a quantitative risk assessment since adequate mitigating equipment remained available. The finding has a cross-cutting aspect in the area of human performance associated with the decision-making component

because the licensee did not ensure that the proposed action was safe in order to proceed, rather than unsafe in order to disapprove the action [H.1(b)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records," associated with the licensee's failure to furnish evidence of an activity affecting quality associated with the 480 Vac breakers. Specifically, the licensee failed to maintain design documents that detailed the correct Digital Low Resistance Ohm (DLRO) values required for ensuring proper connections between the Square D Masterpact NW breaker/cradle assembly to the GE AKD-5, 480 Vac cubicle stabs. The licensee re-generated acceptance criteria to address this issue. This issue was entered into the licensee's corrective action program as CR 2013-04032.

  The licensee's failure to furnish evidence that showed the required DLRO values ensured proper connections between the Square D Masterpact NW breaker/cradle assemble to the GE AKD-5, 480 V cubicle stabs is a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it affected the design control attribute of the Mitigating Systems Cornerstone, and it directly affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding had a cross-cutting aspect in the area of human performance, associated with the resources component, because the licensee failed to maintain complete, accurate, and up-to-date design documentation. Specifically, the licensee did not maintain the engineering process for determining acceptable DLRO values [H.2(c)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the licensee's approval of Root Cause Analysis 2013-03424, Revision 0 and Revision 1, "MSPI Safety System Functional Failures Degrading Trend," which did not assure corrective actions to prevent repetition of a significant condition adverse to quality. The licensee's addressed this issue by revising the root cause analysis. The licensee entered this deficiency into their corrective action program for resolution as CRs 2013-00584 and 2013-14614.

The licensee's failure to establish measures to assure that the cause of the degrading trend in MSPI safety system functional failures would be promptly identified and action taken to preclude repetition in accordance with 10 CFR Part 50, Appendix B, Criterion XVI, was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because the failure to correct the cause and preclude the repetition of the cause would have the potential to lead to a more significant safety concern. Specifically, failure to identify the correct cause and preclude repetition could lead to a high frequency of safety system functional failures. This finding was associated with the mitigating systems cornerstone. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to be of very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the problem and, consequently, the resolution did not identify the extent of cause as necessary [P.1(c)](Section 4OA4).

- <u>Green</u>. The team identified multiple examples of a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to control deviations from design standards. Specifically, the licensee failed to control deviations from the design basis requirements for structural calculations related to the reactor coolant system. The licensee took action to perform additional analysis to confirm the operability of the affected components and to determine the scope of the problem. The licensee entered this deficiency into their corrective action program for resolution as CRs 2013-19878, 2013-18361, 2013-20281, and 2013-14726.

  The failure to control deviations from quality standards as required by 10 CFR Part 50, Appendix B, Criterion III, was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the design control attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and

qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. There was no cross-cutting aspect assigned to this finding because this issue does not reflect present licensee performance (Section 4OA4).

- Green. The team identified multiple examples of a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee's failed to follow station procedures for corrective actions, operability evaluations, and performance of calculations for instances where the licensee's interim operability procedure was invoked for degraded conditions associated with piping and pipe supports. As a result, non-conservative design inputs were used without entering the non-conformances into the corrective action process or performing procedurally required operability evaluations. The licensee's corrective action was to capture the identified instances in the corrective action program and discontinue the use of the interim operability procedure. This issue was entered into the licensee's corrective action program as CR 2013-03598.

  The failure to follow the interim operability procedure was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the human performance attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, and guidance from the Office of Nuclear Reactor Regulation, Division of Engineering technical staff for issues where the inputs to calculations deviated from approved standards, the finding was determined to have very low safety significance (Green) because: (1) the Office of Nuclear Reactor Regulation technical staff determined the non-conformances would not render the evaluated component as inoperable or unable to perform its safety function"; (2) it was not a deficiency affecting the design and qualification of a mitigating structure, system, or component; and (3) it did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of human performance associated with work practices component because the licensee failed to define and effectively communicate expectations regarding compliance with station procedures [H.4(b)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the licensee’s failure to correct conditions adverse to quality in safety-related equipment. The team identified multiple examples where an interim operability criteria procedure was applied instead of correcting the conditions adverse to quality in a timely manner. The licensee’s corrective actions included performing an extent of condition review to identify similar issues and ensure they are entered into the corrective action program for appropriate resolution. This issue was entered into the licensee’s corrective action program as CR 2013-22426.

  The failure to correct conditions adverse to quality is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee’s maintenance rule program. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee had failed to implement a corrective action program with a low threshold for identifying issues to ensure that an issue potentially affecting nuclear safety was promptly identified and fully evaluated [P.1(a)](Section 4OA4).

- Green. The team identified a non-cited violation of Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to develop an adequate procedure for assessing operability of degraded piping and pipe supports. Specifically, Station Procedure PED-MEI-17, "Interim Operability Criteria," a procedure the licensee used to evaluate CQE and L-CQE piping and piping supports that are found to exceed design basis requirements, was inadequate for this application because it did not contain all applicable constraints. The licensee’s corrective actions were to capture the identified instances in the corrective action program and discontinue the use of the interim operability procedure. This issue was entered into the licensee’s corrective action program as CR 2013-22342.

  The failure to use an adequate procedure for evaluating degraded or nonconforming pipe and pipe supports is a performance deficiency. This

performance deficiency was more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, and guidance from the Office of Nuclear Reactor Regulation, Division of Engineering technical staff for issues where the inputs to calculations deviated from approved standards, the finding was determined to have very low safety significance (Green) because: (1) the Office of Nuclear Reactor Regulation technical staff determined the non-conformances would not render the evaluated component as inoperable or unable to perform its safety function"; (2) it was not a deficiency affecting the design and qualification of a mitigating structure, system, or component; and (3) it did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. There was no cross-cutting aspect assigned to this finding because this issue does not reflect present licensee performance (Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to follow Station Procedure NOD-QP-31, "Operability Determination Process." Specifically, Step 4.3.15 required, in part, that, "A positive determination of operability must be justified, including … a technical discussion of why the concern identified does not prevent the item from fulfilling its intended safety function." The team identified that the operability determination associated with a component identified as beyond its specified service life lacked adequate technical justification for why the item was operable with the degraded or nonconforming condition. The licensee addressed this issue by establishing an adequate basis for operability for the non-conformances. The licensee entered this deficiency into their corrective action program for resolution as CR 2013-12255.

  The failure to properly assess and document the basis for operability when a degraded or nonconforming condition was identified is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Since the finding involving inadequate operability determinations occurred while in a shutdown condition, the team used Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," and determined the finding to have very low safety significance (Green) because the finding: (1) did not increase the likelihood of a loss of reactor coolant system inventory; (2) did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory

when needed; and (3) did not degrade the licensee's ability to recover decay heat removal once it was lost. This finding has a cross-cutting aspect in the area of human performance, associated with the decision-making component, because the licensee failed to use conservative assumptions in decision making when performing operability determinations [H.1(b)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to conduct an adequate evaluation of the impacts of modifying the turbine driven auxiliary feedwater pump (FW-10) during all modes of operation. Specifically, the licensee instituted an engineering change package to modify the pump from a variable speed to a constant speed setting and did not consider the dynamic system changes that could affect the pump operation for all design basis events and operating conditions. The licensee adequately addressed this issue by performing a detailed analysis that determined the change did not adversely affect the function of the pump. The licensee entered this deficiency into their corrective action program for resolution as CR 2013-10465.

  The failure to evaluate the effects of modifying the turbine driven auxiliary feedwater pump from a variable speed to a constant speed for all modes of operation was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the configuration control attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of human performance associated with the decision-making component because the licensee failed to use conservative assumptions in decision making. Specifically, the licensee did not reanalyze the pump performance parameters to identify any potentially adverse effects of changing the pump to a constant speed control [H.1(b)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's programmatic failure to conduct adequate operating experience reviews for root cause evaluations in accordance with Station Procedure FCSG-24-4, "Condition

Report and Root Cause Evaluation," Revision 5. Specifically, during the course of the inspection, the team identified four specific examples where licensee staff failed to conduct a thorough operating experience review while performing a root cause analysis to determine whether the same or similar problems have occurred at the Fort Calhoun Station or within the industry. Thorough operating experience reviews are important for the identification of corrective actions that prevent the issues from recurring and determining the associated extent of condition and/or generic implications. This issue was entered into the licensee's corrective action program as CR 2013-14205.

The licensee's failure to conduct adequate operating experience reviews for root cause evaluations was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because if left uncorrected it has the potential to lead to a more significant safety concern. Specifically, if the licensee does not thoroughly evaluate operating experience to determine whether the same or similar problems have occurred at the Fort Calhoun Station or within the industry, then effective corrective actions to prevent the issues from recurring may not be implemented and an adequate extent of condition and/or generic implications from the issue may not be identified. This finding was associated with the Mitigating Systems Cornerstone. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 4, "PWR Refueling Operation: RCS level >23' or PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, this finding was determined to be of very low safety significance (Green) because the finding did not require a quantitative risk assessment because adequate mitigating equipment remained available. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the Operating Experience component because the licensee did not use operating experience information, including vendor recommendations and internally generated lessons learned, to support plant safety by implementing and institutionalizing operating experience through changes to station processes, procedures, equipment, and training programs [P.2(b)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to fully incorporate applicable design requirements into the plant design. Specifically, since initial construction the licensee has failed to incorporate a ventilation system for the vital switchgear rooms that was capable of maintaining room temperature within design requirements under all design conditions. This issue does not represent an immediate safety concern because the licensee has compensatory measures in place to maintain room temperatures while corrective actions to resolve the issue are being implemented. This issue was entered into the licensee's corrective action program as CR 2013 9804.

  The failure to fully incorporate applicable design requirements is a performance deficiency. The performance deficiency was determined to be more than minor,

and therefore a finding, because it affected the design control attribute of the Mitigating Systems Cornerstone, and it directly affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the problem and, consequently, the resolution did not identify the extent of cause as necessary [P.1(c)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action," for the licensee's failure to take adequate corrective actions regarding non-Category I (seismic) piping in the intake structure raw water vault. The licensee's corrective actions for this issue involved isolating and removing the piping. The licensee entered this deficiency into their corrective action program for resolution as CRs 2013-04782, 2013-04956, 2013-09256, 2013-10626, and 2013-22090.

  The failure to take adequate corrective action regarding non-Category I (seismic) piping in the intake structure raw water vault is a performance deficiency. The performance deficiency was more than minor, and therefore a finding, as it is associated with the design control attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated July 1, 2012, this finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. The

finding has a cross-cutting aspect in the area of human performance associated with the decision-making component such that the licensee demonstrates that nuclear safety is an overriding priority. Specifically, that the licensee uses conservative assumptions in decision making and adopts 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)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” associated with the licensee’s failure to follow Station Procedure NOD-QP-31, “Operability Determination Process,” to adequately assess and document the basis for operability when a nonconforming condition was identified. Specifically, the licensee did not determine the effect of a ruptured 6-inch pipe in the raw water system with respect to the safety function provided by the raw water system during a design seismic event. To address this issue the licensee revised the operability evaluation and established a reasonable basis for operability. The licensee entered this deficiency into their corrective action program for resolution as CRs 2013-13410 and 2013-13634.

  The failure to adequately assess and document the basis for operability of the raw water system with respect to the non-conforming seismic design criteria is a performance deficiency. The performance deficiency was more than minor, and therefore a finding, as it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” dated July 1, 2012, this finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee’s maintenance rule program. This finding has a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the problem such that the resolutions address causes and extent of conditions. This includes properly classifying, prioritizing, and evaluating for operability and reportability conditions adverse to quality [P.1(c)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” involving the licensee’s

failure to follow procedures when evaluating the flooding mitigation impact of the removal of the motor for raw water Pump B. Specifically, on June 18, 2013, the operability determination for Corrective Action 018 of CR 2011-10302 was not performed in accordance with Station Procedure NOD-QP-31, "Operability Determination Process," Step 4.3.15, and consequently, failed to evaluate the impact of having only two diversely powered available raw water pumps to support shutdown cooling system operability during a postulated site flood. This issue did not represent an immediate safety concern and has been entered into the corrective action program as CR 2013-15270.

The failure to properly assess and document the basis for operability when a degraded or nonconforming condition was identified is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 4, "PWR Refueling Operation: RCS level >23' or PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, this finding was determined to be of very low safety significance (Green) because the finding did not require a quantitative risk assessment because adequate mitigating equipment remained available. This finding has a cross-cutting aspect in the area of human performance associated with the work control component. Specifically, the team identified that the licensee failed to adequately plan and coordinate work activities, in which, interdepartmental coordination was necessary to assure plant and human performance [H.3(b)](Section 4OA4).

- <u>Green</u>. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to correctly translate the acceptance limit of intake sluice gate leakage values into procedures. Specifically, the acceptance limit from the licensee's testing was applied to 1000 feet of intake level and not to the 983 to 988 feet operating band prescribed in Section I – Flooding, of Station Procedure AOP-01, "Acts of Nature." This issue did not represent an immediate safety concern and has been entered into the corrective action program as CR 2013-15287.

  The failure to fully incorporate applicable design requirements is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it is associated with the design control attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 4, "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the

Pressurizer," dated May 25, 2004, the team determined that because this finding did not increase the likelihood of a loss of reactor coolant system inventory; did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory; and did not degrade the licensee's ability to recover decay heat removal. This finding did not require a Phase 2 or 3 analysis as stated in Checklist 4. Therefore, the finding is determined to have very low safety significance (Green). This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate problems such that the resolutions address causes and extent of conditions [P.1(c)](Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to maintain an adequate procedure for site flooding. Specifically, since June 2013the licensee failed to include appropriate quantitative or qualitative acceptance criteria for Section I – Flooding, of Station Procedure AOP-01, "Acts of Nature," on how to proceed if steps taken to maintain intake cell level less than 988 feet were unsuccessful during a flooding event. This issue did not represent an immediate safety concern and has been entered into the corrective action program as CR 2013-15289.

  The licensee's failure to maintain an adequate procedure for maintaining intake cell level during a flood is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 4, "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, the finding is determined to have very low safety significance (Green) because the finding did not increase the likelihood of a loss of reactor coolant system inventory; did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory; and did not degrade the licensee's ability to recover decay heat removal. This finding did not require a Phase 2 or 3 analysis as stated in Checklist 4. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate problems such that the resolutions address causes and extent of conditions [P.1(c)](Section 4OA4).

- Green. The team identified a non-cited violation of License Condition 3.D, "Fire Protection Program," for the failure to translate Appendix R license exemptions into the fire protection program design. Specifically, the licensee failed to translate the exemption from 10 CFR Part 50, Appendix R, Section III.G, that was

granted July 3, 1985, for the Intake Structure, Fire Area 31, into a design that met those exemptions. The licensee did not protect the cables for both raw water pumps AC-10A and AC-10B from any credible fire in the intake structure. This issue did not represent an immediate safety concern and was entered into the licensee's corrective action program as CR 2013-16201.

The failure to translate Appendix R license exemptions into the fire protection program design is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the protection against external factors attribute of the Mitigating Systems Cornerstone and affected the associated objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," dated September 20, 2013, Step 1.3, the team determined that the reactor would have been able to reach and maintain cold shutdown, therefore, this finding was determined to have very low safety significance (Green). There was no cross-cutting aspect assigned to this finding because the deficiency was over three years ago and does not reflect present licensee performance (Section 4OA4).

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to document the extent of condition review for a number of Root Cause Analyses in accordance with corrective action program procedures. Specifically, during the course of the inspection, the team identified four examples where the licensee did not follow Station Procedure FCSG-24-4, "Condition Report and Cause Evaluation," and, as a result, did not evaluate the extent to which the actual conditions existed with other plant processes, systems, equipment, or human performance related activities. This issue does not represent an immediate safety concern and was entered into their corrective action program as condition report CR 2013-18291.

  The failure to follow the requirements of Station Procedure FCSG-24-4 when documenting extent of condition reviews in multiple Root Cause Analyses was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because if left uncorrected the failure to perform extent of condition reviews could lead to a more significant safety concern. Specifically, the failure to identify and address additional conditions adverse to quality in the extent of condition review has the potential to lead to a failure to recognize degraded equipment in a timely manner. This finding was associated with the Mitigating Systems Cornerstone. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 4, "PWR Refueling Operation: RCS level >23' or PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, the team determined that this finding was of very low safety significance (Green) because the finding did not require a quantitative risk assessment because adequate mitigating equipment remained available. The

team determined this finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee failed to thoroughly evaluate problems such that the resolutions address the causes [P.1(c)](Section 4OA4).

Other Findings

- Severity Level IV. The team identified a non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," associated with the licensee's failure to adequately evaluate Modification EC 33464, "Replace AK-50 480 V Main and Bus-Tie Breakers With Molded Case Type or Equivalent," to determine if it required prior NRC approval. Specifically, the licensee's documented evaluation failed to identify and evaluate new creditable failure modes to determine whether they would have an adverse effect on the 480 Vac electrical distribution system. The licensee's corrective action was to revise the evaluation. This issue was entered into the licensee's corrective action program as CR 2013-04474 and 2013-16954.

  The licensee's failure to implement the requirements of 10 CFR 50.59 and adequately evaluate changes associated with the electrical distribution system is a performance deficiency. Because this performance deficiency had the potential to impact the NRC's ability to perform its regulatory function, the team evaluated the performance deficiency using traditional enforcement. In accordance with Section 2.1.3.E.6 of the NRC Enforcement Manual, the team evaluated this finding using the significance determination process to assess its significance. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the finding is determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design or qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its Technical Specification allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program; and (5) did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding, or severe weather event. Therefore, in accordance with Section 6.1.d.2 of the NRC Enforcement Policy, the team characterized this performance deficiency as a Severity Level IV violation. The team determined that a cross-cutting aspect was not applicable to this performance deficiency because the failure to adequately evaluate changes in accordance with 10 CFR 50.59 was strictly associated with a traditional enforcement violation (Section 4OA4).

- Severity Level IV. The team identified three examples of a Severity Level IV non-cited violation of 10 CFR 50.73, "Immediate Notification Requirements for Operating Nuclear Power Reactors," associated with the licensee's failure to submit a Licensee Event Report within 60 days following a discovery of an event

meeting the reportability criteria as specified. The licensee's corrective actions were to submit the licensee event reports. The licensee entered this deficiency into their corrective action program for resolution as CRs 2013-12863 and 2012-03796.

The team determined that the failure to make a required Licensee Event Report is a violation of 10 CFR 50.73. The violation was evaluated using Section 2.2.4 of the NRC Enforcement Policy because the failure to submit a required licensee event report may impact the ability of the NRC to perform its regulatory oversight function. As a result, this violation was evaluated using traditional enforcement. In accordance with Section 6.9 of the NRC Enforcement Policy, this violation was determined to be a Severity Level IV non-cited violation. The team determined that a cross-cutting aspect was not applicable to this performance deficiency because the failure to make a required report was strictly associated with a traditional enforcement violation (Section 4OA4).

- Severity Level IV. The team identified a cited Severity Level IV violation of 10 CFR 50.9, "Complete and Accurate Information," and an associated reactor oversight program finding (NCV 05000285/2013013-19, "Failure to Translate Appendix R License Exemptions into the Plants Fire Protection Program Design"), for the licensee's failure to provide information to the Commission that was complete and accurate in all material respects. Specifically, when responding to a request for additional information, the licensee supplied incorrect information to the NRC and this information was subsequently used by the NRC to support a license amendment for the station. This issue was entered into the station's corrective action program as CR 2013-15021.

  The failure to provide the NRC with complete and accurate information when responding to a request for additional information is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, "Issue Screening," Figure 1, dated September 7, 2012, the team determined that the failure to provide complete and accurate information was a performance deficiency that required evaluation under both traditional enforcement and the reactor oversight program. The performance deficiency was determined to be more than minor because: (1) the information was considered material to the NRC's decision making process; and (2) it affected the equipment performance attribute of the Mitigating Systems Cornerstone with regard to availability, reliability, and capability of the raw water pumps to perform their safety function during a fire in the intake structure. Using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the team determined the finding to have very low safety significance (Green) because it only affected the ability to reach and maintain cold shutdown conditions. Under the traditional enforcement review, the team determined that in accordance with Section 6.9.c.1 of the NRC Enforcement Policy, this finding represented a Severity Level III violation. Specifically, the team determined that if this information had been completely and accurately provided, it would likely have caused the NRC to undertake a substantial further inquiry. The NRC takes the issue of complete and accurate

license submittals very seriously. For this reason, the NRC considered citing this as a Severity Level III violation, as discussed in the Enforcement Policy, as the NRC had approved a licensing action based on the incorrect information. However, after consideration by NRC management, and with the approval of the Director of the Office of Enforcement, it was determined that a Severity Level IV, cited violation was appropriate. This decision was based on the very low safety significance (Green) of the associated reactor oversight program finding (05000285/2013013-19). There was no cross-cutting aspect assigned to this finding because the inaccurate information was provided over three years ago and this issue does not reflect present licensee performance (Section 4OA4).

B. Licensee-Identified Violations

None.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA4 IMC 0350 Inspection Activities (92702)

The inspection team continued the NRC Inspection Manual Chapter 0350 inspection activities, which included follow-up on the Restart Checklist contained in Confirmatory Action Letter (CAL) EA-13-020 issued February 26, 2013. The purpose of this inspection was to perform an assessment of the causes of the performance decline at the Fort Calhoun Station (FCS), to assess whether planned corrective actions are sufficient to address the root causes and contributing causes and to prevent their recurrence, and to verify that adequate qualitative or quantitative measures for determining the effectiveness of the corrective actions are in place. These assessments were used by the NRC to independently determine if plant personnel, equipment, and processes were ready to support the safe restart and continued safe operation of the Fort Calhoun Station.

The team used the criteria described in baseline and supplemental inspection procedures, various programmatic NRC inspection procedures, and Inspection Manual Chapter 0350 to assess Omaha Public Power District's (the licensee) performance and progress in implementing its performance improvement initiatives. The team 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 7, 2013, through February 18, 2014. Specific documents reviewed during this inspection are listed in the attachment.

The following inspection scope, observations and findings, and assessments, are documented by the Confirmatory Action Letter Restart Checklist (CL) item number.

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

Section 1 of the Restart Checklist contains those items necessary to develop a comprehensive understanding of the root causes of safety-significant performance deficiencies identified at the 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 5.

Item 1.c: Electrical Bus Modification and Maintenance – Red Finding

(1) Inspection Scope

a. The team assessed the licensee's actions taken since inspection activities documented in NRC Inspection Report 05000285/2013008. As documented in Inspection Report 05000285/2013008, the team reviewed this area for closure and noted discrepancies which lead to area 1.c being left open. The team reviewed the licensee's actions to address the teams' concerns to ascertain whether they were sufficient to ensure plant safety and support closure of the restart checklist items associated with the Red finding and notice of violation issued to the licensee on April 10, 2012.

   The team assessed the root cause analyses the licensee developed and included in its closure book for the Red finding (i.e., Closure Book 1.C): RCA 2011-05414, "Breaker Cubicle 1B4A Fire," Revision 3, dated October 5, 2012, and RCA 2011-06621, "1B3A Main Breaker Trip During Switchgear Fault on 1B4A," dated May 3, 2012. The focus of RCA 2011-05414 was identifying the conditions surrounding the initiation of the fire event that occurred on June 7, 2011, and determining what created the fire and subsequent loss of 480 Vac, Bus 1B4A. The purpose of RCA 2011-06621 was to determine why an adequate level of separation between two trains of 480 Vac power was not maintained during the fire event; however, the purpose statement was redefined several times throughout the document.

   The team's assessment was based on the following objectives:

   - Provide assurance that the root and contributing causes of risk-significant issues were understood

   - Provide assurance that the extent-of-condition and extent-of-cause of risk-significant issues were identified

   - Provide assurance that the licensee's corrective actions for risk-significant performance issues were, or will be, sufficient to address the root and contributing causes and to preclude repetition

b. Open items (Licensee Event Reports and Violations), specifically related to the Red finding were reviewed by the team. The team verified the adequacy of the licensee's causal analysis and extent of condition evaluations related to and associated with the Red finding. In addition, the team verified that adequate corrective actions were identified and associated with the licensee's root and contributing causes and extent of condition evaluations, and that these corrective actions are either implemented or appropriately scheduled for implementation.

(2) Observations and Findings

a. <u>Licensee's Assessment of the Red finding</u>

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

The team determined that the licensee had evaluated the issue using systematic methodologies to identify root and contributing causes. Specifically, Root Cause Analysis (RCA) 2011-06621, Revision 2, stated that the analytical methods used during the investigation included events and causal factors charting and fault tree analysis. A fault tree was created for the event in an attempt to identify all possible means by which load center 1B3A main feeder breaker could have opened inappropriately given the circumstances. The root cause analysis stated that the fault tree analysis was essentially a failure modes and effects analysis which identified: (1) human performance; (2) programmatic; and (3) oversight factors which were considered to finally arrive at the root cause. The root cause analysis contained the fault tree analysis created for this investigation.

RCA 2011-06621, Revision 2, documented the following root and contributing causes of inadequate separation of safety-related equipment:

- Root Cause-1 (8.1): Deleted – see contributing cause-4 (8.6).

- Root Cause-2 (8.2): Design Change Package preparation procedures do not provide guidance to evaluate design features of new components in regard to the possibility that they may have adversely affected required performance characteristics if not properly configured.

- Contributing Cause-1 (8.3): Detailed standards for performing and documenting wire/continuity checks for new wiring do not exist. It is left to the test and field engineer to judge the level of detail required.

- Contributing Cause-2 (8.4): The design engineer did not properly employ the human performance toolbox in regard to maintaining a questioning attitude about the details of operation of new breakers.

- Contributing Cause-3 (8.5): The field engineer and electricians did not properly employ the human performance toolbox in that they did not question the lack of detail in the Construction Work Order for performing wire and continuity checks.

- Contributing Cause-4 (8.6): The vendor manual for the Masterpact breakers does not clearly state how the Zone Select Interlock, if not properly restrained, will impact breaker coordination. The vendor was unaware of the effect of the Full Function Test Kit on the Zone Select Interlock functionality. This knowledge gap resulted in a failure to specify a functional test that would ensure proper breaker performance. The knowledge gap is also being

investigated by the vendor. (Refer to NLI NCR number 410. Note: Any root or contributing cause associated with vendor actions will be addressed by the vendor's corrective action program and not by OPPD's program.)

The team determined that in RCA 2011-06621, Revision 2, the licensee had adequately used systematic methodologies to identify the root and contributing causes for the failure to maintain separation between two trains of 480 Vac power during the fire event. The team noted that the licensee had deleted Root Cause 1 (8.1) and made it part of Contributing Cause 4 (8.6). This addresses the concerns identified by NRC Inspection Reports 05000285/2012004 and 05000285/2013008 with respect to Root Cause 1 (8.1).

<u>Determine that the root cause evaluation address the extent of condition and the extent of cause of the problem.</u>

RCA 2011-06621 defined the condition as the failure to properly disable the Zone Select Interlock breaker feature which resulted in a loss of expected coordination between adjacent 480 Vac breakers. During the June 7, 2011, fire event, the failure to restrain the Zone Select Interlock was caused by a wiring error, which occurred during installation of the restraining jumpers. The licensee identified other conditions that could cause the Zone Select Interlock not to be adequately restrained, including snap-in connectors not firmly mounted and popping out during breaker racking and a damaged mounting-bracket linkage arm that could cause incomplete circuits at the input of the breaker. The licensee determined that the extent of condition was the possibility that any or all these failure modes could exist on any of the twelve Masterpact NW breakers installed in the 480 Vac switchgear. The Zone Select Interlock wires were checked at all twelve breakers and cradles. In the course of the root cause analysis, other adverse breaker conditions were identified and checked. Closure Book 1.c stated that the licensee has verified the correct placement and continuity of the other Zone Select Interlocks jumpers in the station and was verifying breaker overcurrent coordination through primary injection testing without using a Full Function Test Kit. The licensee implemented new guidance for testing control wiring that is applicable to all modified and maintained electrical circuits. This was accomplished in condition report action items 2011-06621-28 and 2011-06621-32. The team determined that the licensee had failed to promptly identify and correct a condition adverse to quality. Specifically, the team reviewed the licensee's corrective actions and determined that action item 2011-06621-32 had not been performed, but had been identified as complete and was closed due to an administrative error. The team identified this performance deficiency as, NCV 05000285/2013013-01, "Failure to Complete all Testing for a Condition Adverse to Quality," which is further discussed in Section 5 of this report.

RCA 2011-06621, Revision 2, identified the root cause as the lack of specific direction in the Design Change Package preparation procedure to require the design engineer to consider the impact of design features of new equipment if not properly disabled. The root cause analysis stated, "An extent of cause is other electrical modifications susceptible to a lack of appropriate consideration of new failure modes that could exist because new design features are not properly disabled." The closure

book stated that the root cause has been corrected by revising the appropriate design procedures for all engineering disciplines to require a comparison of new features with the original equipment including a consideration of critical parameters within the design change process. The licensee implemented corrective actions to review other electrical/I&C modifications from the last five years to determine if failure modes introduced by features not part of the original equipment could have been introduced.

<u>Determine that the root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in IMC0310.</u>

The safety culture analysis portion of the root cause analysis failed to identify the reasons for why some safety culture aspects were not applicable, as required by station procedure. This information was important for complete understanding of the circumstances surrounding the event, and to ensure that other root and contributing causes were not inappropriately ruled out. The form for documenting the safety culture analysis was not consistent with the instructions in the governing procedure with respect to documenting the reasons why a safety culture aspect was not applicable. The form required the licensee to bin the root cause and contributing causes into the various components, which would not provide an opportunity to determine if the causal analysis failed to identify other root and contributing causes.

<u>Determine that appropriate corrective actions are specified for each root and contributing cause.</u>

Corrective action items and schedules for implementing these items were specified for the root and contributing causes discussed in RCA 2011-06221. Closure Book 1.c provided a table that outlined which corrective actions correlated to various causes. The team determined that these corrective actions were adequate to address those causes.

During their review the team determined that the licensee had failed to provide an appropriate calculation to establish the basis for testing of safety related breakers. The team identified this performance deficiency as NCV 05000285/2013013-02, “Failure to Furnish Evidence of an Activity Affecting Quality.” The team also determined that the licensee had performed an inadequate 10 CFR 50.59 evaluation for modifications performed on safety related breakers. The team identified this performance deficiency as NCV 05000285/2013013-03, “Failure to Evaluate Changes to Ensure They Did Not Require Prior Approval.” These issues are further discussed in Section 5 of this report.

<u>Determine that a schedule has been established for implementing and completing the corrective actions.</u>

Corrective action items and schedules for implementing these items were specified for the root and contributing causes discussed in RCA 2011-06621. Remaining corrective actions were discussed in the previous sections of this report. The team did not identify any issues associated with licensee’s schedule.

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

RCA 2011-06621, Revision 2, does not address the concern identified in NRC Inspection Report 05000285/2013008. Specifically, because a procedural correction may not be effective in precluding repetition of events, the licensee should have established more frequent effectiveness reviews for the procedural corrective actions. This effectiveness review has acceptable acceptance criteria (i.e., no issues in form, fit, or function); however, the team determined that the corrective actions need more run-time and interim effectiveness reviews in accordance with Procedure FCSG-24-5, "Cause Evaluation Manual," Revision 5 before a conclusion can be made about their effectiveness.

b. Resolution of Open Items Related to the Red Finding

The team reviewed the following open items:

LER 2011010-01	Fire Causes a Circuit Breaker to Open Outside Design Assumptions
VIO 2012010-01	Failure to Ensure that the 480 VAC Electrical Power Distribution System Design Requirements were Implemented and Maintained
VIO 2012007-02	Failure to Maintain Command and Control Function During Fire Fighting Activities in the Protected Area
VIO 2012004-04	Failure to Ensure Breaker Coordination of 480 Vac Electrical Power Distribution System Was Maintained

The team verified the adequacy of the licensee's causal analyses and extent of condition evaluations. In addition, the team verified that adequate corrective actions were identified and associated with the licensee's root and contributing causes and extent of condition evaluations, and that, implementation of these corrective actions are either implemented or appropriately scheduled for implementation.

During this review, the team determined that the licensee had failed to make a required licensee event report to the NRC. The team identified this performance deficiency as NCV 05000285/2013013-04, "Failure to Submit Licensee Event Report," which is further discussed in Section 5 of this report.

(3) Assessment Results

a. The team has concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews, that this area was adequately addressed by the licensee and the following Restart Checklist Items are closed:

1.c.1	Electrical Fire Red Finding root and contributing cause evaluation
1.c.2	Electrical Fire Red Finding extent-of-condition and cause evaluation
1.c.3	Electrical Fire Red Finding corrective actions addressing root and contributing causes
1.3.1.1	Rebuild the 1B4A load center
1.3.1.2	Provide documentation for the dedication of the rebuilt load center in accordance with Contract 163495
1.3.1.3	Complete Engineering Change 53257 and obtain PRC approval to authorize the use of the rebuilt load center, 1B4A
1.3.1.7	Complete Engineering Change 53517 that details the repair to the cable jackets for cables located in the cable tray above 1B4A load center
1.3.1.8	Repair or replace the cables located in the cable tray above load center 1B4A that have had jacket damage
1.3.1.10	Calibration of the internal relays and protection equipment for Bus 1B4A
1.3.1.12	Calibrate new Square D circuit breakers
1.3.1.17	Perform testing of all circuits associated with 1B4A load center
1.3.1.19	Submit, track, and seek approval of procedures that are changed as the result of EC 53257 and are required to be issued before the System Acceptance Process.
1.3.1.21	Declare Bus 1B4A Operable
1.3.1.23	Extent-of-condition repair requirements. Provide repair requirements for extent-of-condition.
1.3.1.24	Implement the requirements supplied by System Engineering regarding the extent-of-condition.

LER 2012010-01	Fire Causes a Circuit Breaker to Open Outside Design Assumptions
VIO 2012010-01	Failure to Ensure that the 480 Vac Electrical Power Distribution System Design Requirements were Implemented and Maintained
VIO 2012007-02	Failure to Maintain Command and Control Function During Fire Fighting Activities in the Protected Area
VIO 2012004-04	Failure to Ensure Breaker Coordination of 480 Vac Electrical Power Distribution System Was Maintained

Item 1.g: Safety System Functional Failures White Performance Indicator

(1) Inspection Scope

The team reviewed the licensee's programmatic evaluation associated with safety system functional failures, as well as the cause evaluations associated with the individual licensee event reports identified in Area 1.g of Restart Checklist Basis Document, Revision 4. The purpose of these reviews was to independently verify that the licensee had performed adequate casual analyses and extent of condition evaluations related to these issues. In addition, the team verified that adequate corrective actions were identified and associated with the causes and extent of condition evaluations, and that, implementation of these corrective actions were either implemented or appropriately scheduled for implementation.

(2) Observations and Findings

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

The team determined that the licensee evaluated the condition using systematic methodologies and problem analysis techniques to identify the root and contributing causes. The licensee used the following systematic methods to complete the root cause analysis: (1) event and causal factors charting to allow complex issues to be organized to clearly identify the structure of the event and its cause; and (2) common factors analysis to understand the major common issues that factored into the Mitigating Systems Performance Indicator (MSPI) degradation.

The team concluded that the use of the techniques provided an adequate methodology for evaluating the problem.

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

The team determined that the root cause evaluation was appropriately conducted to a level of detail commensurate with a Significance Level 1 event or condition – An event or condition that is a Significant Condition Adverse to Quality that has major potential or actual impact. The event presents significant risk or consequences to the safe, reliable operation of the plant, personnel safety, or organizational and human behaviors, such that, recurrence is unacceptable – in accordance with Licensee Procedure FCSG-24-3, "Condition Report Screening," Revision 7.

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

The team determined that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience, as required, by Station Procedure FCSG-24-4, "Condition Report and Cause Evaluation," Revision 7.

<u>Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem.</u>

The team determined the evaluation of the extent of condition was not complete. RCA 2013-03424 determined the bounding condition as:

> "MSPI Safety System Functional Failure indicator degrading trend (increase in LER submittals due to Safety System Functional Failures as a result of discovering latent design basis/configuration control issues)."

The root cause concluded that an extent of condition exists, and that, this condition has been repeatedly identified as design/configuration control anomalies. The root cause also concluded that any processes which rely upon clear and accurate design basis could be impacted by latent undiscovered design anomalies.

The root cause acknowledged that the condition could extend to other processes and programs, such as, fuel loading analysis, surveillance testing, preventative maintenance, and equipment qualification; however, it did not determine to what extent the actual processes and programs were affected. This is contrary to Station Procedure FCSG-24-4, Attachment 1, Section F, "Extent of Condition," Paragraph 1.2, which states, in part, "The extent to which the actual condition of the Problem Statement exists in other applicable plant processes, systems, equipment, or human performance related activities (programs) SHALL be determined."

In interviews with licensee personnel, the team was told that the extent of condition review was scheduled for a later date because the depth of review would be large and corrective actions in CRs 2012-08134 and 2012-02857 would address some of the programs already mentioned. Delaying the extent of condition review is allowed by Station Procedure FCSG-24-5, "Cause Evaluation Manual," when the investigator and condition report owner may exercise conservative judgment to determine how deep to pursue the extent of condition. However, if the full scope or impact is to be determined later, then the corrective action plan must include one or more supporting actions to do so. Corrective actions to perform the full extent of condition were not included in RCA 2013-03424.

The team determined the evaluation of the extent of cause to be inadequate. RCA 2013-0324 determined the failure, "to maintain an environment, in the Engineering Division, that valued maintaining the license and design basis of the station over continued operation of the facility," to be the root cause of the declining performance indicator. The root cause also established that the potential existed for this cause to further impact other processes within Engineering (e.g. that an extent of cause existed). It did not, however, determine what the extent of cause was, and thus, could not assure that corrective actions would be broad enough to prevent repetition (e.g. another safety system functional failure related to the extent of root cause elsewhere in Engineering or outside of Engineering).

Specifically, the licensee determined the declining performance indicator to be a significant condition adverse to quality (SCAQ), and that, the potential existed both:

(a) for its root cause to impact other processes; and (b) for that cause which triggered behaviors associated with the condition to trigger similar behaviors in other processes (e.g. failure to maintain an environment in other divisions that valued maintaining the license and design basis of the station over continued operation of the facility), but did not determine the actual extent of cause (e.g. in which divisions this cause could repeat and result in or contribute to White Performance Indicator repetition). Instead, the RCA established future tasking actions to determine the extent of cause corrective actions intended to prevent repetition without knowing the actual extent of cause.

The team reviewed the RCA established future tasking actions, intended to determine extent of cause, to determine if they could be relied upon to assure revision to RCA 2013-03424 corrective actions to prevent repetition (CAPRs). The team determined that the actions tasked against RCA 2013-05570 could not be relied upon for at least three reasons. First, the tasking was not directed to any specific element. Secondly, the team's review of RCA 2013-05570 found that it lacked any meaningful linkage back to RCA 2013-03424 to assure that it would provide the specific extent-of-cause information being sought. Finally, the team determined that RCA 2013-05570 was itself, inadequate. As discussed further below, this lack of meaningful linkage also placed at risk the bulk of RCA 2013-03424 corrective actions which, like the extent of cause tasking, were assigned to RCA 2013-05570.

The team informed the licensee of these concerns, and the licensee initiated CR 2013-14584 to capture this issue in the station corrective action program. The licensee revised RCA 2013-03424 to address the issues identified by the team.

In the revised root cause analysis the licensee determined that the identified root cause extended beyond the engineering organization, and had been repeatedly identified as design basis/configuration issues, but actions taken by management to address the dormant nature of the existing design basis issues had limited effectiveness. To address the identified extent of cause the licensee developed corrective actions specified in CR 2013-03424, and linked corrective actions from CR 2013-05570 to CR 2013-03424 in the corrective action program. The team determined that these actions were adequate to identify the extent of cause, and to implement corrective actions to address the extent of cause.

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

The team determined that the root cause evaluations appropriately considered the safety culture components as described in Inspection Manual Chapter 0310. The licensee reviewed each safety culture component and determined if the condition was applicable. Station Procedure FCSG-24-4, "Condition Report and Cause Evaluation," Revision 7, Section L, Paragraph 1.3, states, "For Safety Culture Aspects that are found to be applicable, reference the root and contributing causes

and the specific corrective actions that address that aspect issue." The team determined that the actions were appropriate.

<u>Determine that appropriate corrective actions are specified for each root and contributing cause.</u>

Revision 0 of RCA 2013-03424 originally identified the root cause as, "Fort Calhoun Station engineering management failed to maintain control over the design and configuration of the Fort Calhoun Station." The corrective action to prevent recurrence in Revision 0 of RCA 2013-03424 was documented as:

> "Identify and define the Licensing bases and assure licensing bases documentation remains current, accurate, complete, and retrievable.
>
> - Identification includes determining the record types.
> - Identify a consistent numbering system.
> - Establish methodology (database) for ensuring current and historical licensing bases records are readily retrievable.
> - Reconstitute (identify, locate, and store in a retrievable method) the licensing bases including historical records required to establish the current bases.
> - If conflicts are identified during identification and location of licensing bases documentation, a Condition Report is initiated to document and track the resolution.
> - Establish process for assuring licensing bases documentation remains current, accurate, complete, and retrievable. Current processes may be retained or revised to assure needed results.
> - Closure determination: Conduct an outside independent assessment to validate the completion of identifying all license bases documents are retrievable, and that, the process for updates is implemented."

The team determined that the corrective action to prevent recurrence for the root cause specified in Revision 0 of RCA 2013-03424 was not appropriate and would not prevent recurrence of the root cause. The team determined that the root cause was narrowly focused on the management of the engineering division and failed to identify a culture in the engineering division, as a whole, that failed to maintain the design and configuration control. This condition was captured in CR 2013-12236. The team identified this performance deficiency as NCV 05000285/2013013-05, "Inadequate Corrective Actions to Prevent Repetition of A Significant Condition Adverse to Quality, a White MSPI SSFF Degrading Trend," which is further discussed in Section 5 of this report.

The licensee revised RCA 2013-03424 to include a new root cause and an additional corrective action. Revision 1 of RCA 2013-03424 revised the root cause to, "Fort Calhoun Station failed to maintain an environment, in the Engineering Division, that valued maintaining the license and design basis of the station over continued operation of the facility. This led to a loss of control over the design and configuration of the Fort Calhoun Station." An additional corrective action to prevent recurrence was included to strengthen the function of the oversight group that performs reviews of engineering products.

The team determined that these corrective actions were adequate to address the the identified causes.

<u>Determine that a schedule has been established for implementing and completing the corrective actions.</u>

The team determined that a schedule had been established for implementing and completing the corrective actions. However, the due dates for corrective actions to preclude repetition were not explicitly documented in the corrective action matrix of RCA 2013-03424. Rather, the reader is referred to RCA 2015-05570.

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

Similar to observations above, in which RCA 2013-3424 leveraged RCA 2013-05570 extensively, it also leverage the effectiveness review of that RCA's corrective actions to prevent recurrence of that RCA's root cause. However, because the root causes of RCA 2013-05570 differed substantively from the root cause in RCA 2013-03424, the team determined that the RCA 2013-05570 effectiveness review did not constitute an appropriate measure of success of the corrective actions to prevent recurrence of the RCA 2013-03424 root cause and its extent of cause.

Following revision of RCA 2013-03424 the licensee incorporated adequate effectiveness reviews into this root cause, as well as linking corrective actions from RCA 2013-05570. Specifically, the team noted that RCA 2013-05570 had effectiveness reviews associated with the corrective actions, and by linking the corrective actions from 2013-05570 to 2013-03424 in the corrective action program any identified weaknesses with corrective actions in 2013-05570 would trigger a review under 2013-03424 as well. The team determined this to be adequate.

(3) Assessment Results

The team has concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews, that this area was adequately addressed by the licensee. Restart Checklist Item 1.g is closed.

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

Section 2 of the Restart Checklist contains those items necessary to ensure that important structures, systems, and components affected by the flood and safety significant structures, systems, and components at the Fort Calhoun Station are in appropriate condition to support safe restart and continued safe plant operation.

<u>Item 2.c: Qualification of Containment Electrical Penetrations</u>

(1) Inspection Scope

a. The team reviewed the adequacy of the licensee’s actions associated with the presence of Teflon ® used in a number of containment electrical penetration feedthrough assemblies. Specifically, the team assessed Condition Report CR 2012-1947, for which the “Description” section stated, in part,

> “Test data and analytical techniques demonstrate that FCS feedthrough subassemblies used at FCS containing conductors with Teflon insulation and Teflon seals are susceptible to significant degradation from a postulated Design Basis Event environment.”

The team’s assessment of the licensee’s effectiveness in addressing the deficiency was based on the following criteria:

- Provide assurance that the root and contributing causes of risk-significant issues were understood;
- Provide assurance that the extent-of-condition and extent-of-cause of risk-significant issues were identified;
- Provide assurance that the licensee's corrective actions for risk-significant performance issues were, or will be, sufficient to address the root and contributing causes and to preclude repetition

b. An open item (Licensee Event Report) specifically related to the containment electrical penetration issue was reviewed by the team. The team verified the adequacy of the licensee’s causal analysis and extent of condition evaluation. In addition, the team verified that adequate corrective actions were identified and associated with the licensee’s root and contributing causes and extent of condition evaluations, and that, implementation of these corrective actions are either implemented or appropriately scheduled for implementation.

(2) Observations and Findings

a. Licensee's Assessment of the Containment Penetration Issue

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

The licensee performed a root cause analysis associated with CR 2012-01947 for the condition. The team noted, at the time of the inspection that the licensee had revised the original version of the root cause analysis and the version the team reviewed, was Revision 2, dated July 8, 2013.

The team determined that the licensee evaluated the problem using three systematic methodologies and problem analysis techniques to identify the root and contributing causes. The licensee used the following systematic methods to complete the root cause analysis report: (1) Event and Causal Factors Chart; (2) Barrier Analysis; and (3) Streaming Analysis.

The licensee developed an Event and Causal Factor Chart using historical events to graphically display the timeline of events and factors associated with the events. The licensee then evaluated those events to identify the barriers that could have prevented the condition. From this, the licensee derived the causal factors and performed a streaming analysis on the causal factors to determine which factors were the more fundamental causes that drive the others. Then, the licensee conducted a qualitative evaluation of each causal factor to identify causal factors related to the root cause. The team concluded that the use of these techniques provided an adequate analysis for evaluating the problem.

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

The team determined that the licensee conducted the root cause analysis to a level of detail commensurate with the significance of the problem. The presence of Teflon in containment penetrations represented a potential significant degradation of the containment under accident conditions. The licensee appropriately treated this deficiency as a high level condition in the corrective action process. The licensee identified the following root cause for the condition:

> There was a lack of technical oversight to ensure the information associated with Teflon material used in EQ Containment electrical penetration subassemblies was applied to non-EQ electrical penetrations.

The team considered the identification of this root cause to have been done with an appropriate level of inquiry and depth. The licensee employed their root cause analysis methodology as called for in Procedures NOD-QP-19, "Cause Analysis Program," and FCSG-24-5, "Cause Evaluation Manual."

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

The team determined that the RCA 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 their evaluations. The licensee's search concluded that information was available in the late 1960's that Teflon was not resistant to high radiation levels in reports from Oak Ridge National Laboratory and the Western New York Nuclear Research Center.

The licensee's review of external operating experience identified cases where the Fort Calhoun Station missed opportunities to use operating experience effectively. The licensee identified that few plants used Teflon seals and insulation for containment electrical penetrations, which could have been a missed opportunity to question their practice. The team noted that the licensee did capture this missed opportunity in their corrective action program.

The licensee learned that containment electrical feedthrough subassemblies with a multi-conductor design containing Teflon seals and insulation were only supplied to the Fort Calhoun Station in the United States. In addition, subassemblies with coaxial or triaxial cables with Teflon jackets were only supplied to Salem, Crystal River, and the Fort Calhoun Station. The seals and electrical conductor insulation were made from environmentally qualified material. Based on these reviews, the team concluded the root cause analysis had adequately reviewed operating experience.

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

The team observed that the licensee did, separately and adequately, address both extent of cause and extent of condition. For extent of condition, the licensee considered the extent of condition to be the extent to which the actual condition exists with similar plant processes, equipment, or human performance. Using this, the licensee evaluated the extent of condition (1) the containment personnel air lock electrical penetration subassemblies, which contained Teflon seals and wiring insulation, (2) containment personnel air lock mechanical components, which contained Teflon, and (3) mechanical equipment located in a harsh environment that contained Teflon and performed a containment integrity function. The team confirmed that corrective actions had been generated for these extents of condition and that the actions supported plant safety and restart.

The team also observed that the licensee screened extent of cause to be the extent to which the root cause of an identified problem exists (or may potentially exist) in other plant processes, systems, equipment or human performance related activities. The extent of cause for the root cause was determined to exist in several plant processes, systems, equipment, and human performance related activities. The licensee addressed these in other root cause analyses performed for their performance improvement efforts. These included RCA 2012-08137, "Regulatory

Processes and Infrastructure," RCA 2012-09494, "Deficiencies in Identifying Degraded/Nonconforming Conditions and Performance of Operability Determinations," RCA 2012-08132, "Site Operational Focus," and RCA 2013-02857, "HELB/EEQ not in accordance with 10 CFR 50.49."

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 team determined that the root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in Inspection Manual Chapter 0310. The licensee reviewed each safety culture component and determined if the condition was applicable so that they could link the component to a root or contributing cause.

The safety culture review was aimed at identifying issues with cross-cutting tendencies that warrant enhanced corrective actions to address. Five safety culture aspects were found to be applicable to this root cause. These five cross-cutting aspects were:

- H.1(b) - conservative decision making
- H.2(a) - availability of resources to maintain design margins and minimize long standing issues
- P.1(c) - addressing extent of condition when resolving problems
- P.2(b) – use of operating experience
- O.1(b) – management reinforcing standards and behaviors

The team reviewed that the licensee’s assignment of safety culture aspects and confirmed that the applicable aspects had been addressed by corrective actions.

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 root cause. The licensee specified three corrective actions designated to prevent recurrence. These included integrating leaders having external perspectives and broad experience based insights from external organizations, revising and implementing human performance procedures utilizing best industry practices, and improving the station issue prioritization procedures and processes. Other actions included training on human performance, incorporating current industry best decision making practices, developing and implementing a plan to increase the depth of plant equipment and systems knowledge for engineering personnel, and developing and implementing a plan to increase the depth of licensing and design basis knowledge for engineering personnel. To correct the issue the licensee replaced or capped containment electrical penetrations that used Teflon as electrical insulation or sealant prior to plant startup.

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

The team determined that the Fort Calhoun Station established a schedule for implementing and completing corrective actions. The team noted that CR 2012-01947 and 2010-02387 contained a long list of corrective actions identified to resolve the issue. The team sampled the items to assure that the more risk significant issues were given higher priority. The team concluded that the schedule of corrective actions was adequate.

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

The team determined that the Fort Calhoun Station developed quantitative and qualitative measures of success for determining the effectiveness of the corrective actions to prevent recurrence. These effectiveness reviews were broken down into separate actions in the corrective actions for the root cause analysis. Each of these corrective actions contained detailed means to ascertain the effectiveness measures.

b. The team reviewed the licensee's causal analyses, corrective actions, and extent of condition associated with Licensee Event Report 2012-002, "Inadequate Qualifications for Containment Penetrations Renders Containment Inoperable." In addition, the team verified that adequate corrective actions were identified associated with the causes and extent of condition evaluations and that implementation of these corrective actions were either implemented or appropriately scheduled for implementation.

(3) Assessment Results

a. The team concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews, that this area has been adequately addressed by the licensee. The following restart checklist items for Area 2.c are closed:

2.c.1 Containment electrical penetrations root and contributing cause evaluation

2.c.2 Containment electrical penetrations extent-of-condition and cause evaluation

2.c.3 Containment electrical penetrations corrective actions

b. Licensee Event Report 2012-002, "Inadequate Qualifications for Containment Penetrations Renders Containment Inoperable," will be closed.

3. <u>Adequacy of Significant Programs and Processes</u>

Section 3 of the Restart Checklist addresses major programs and processes in place at the Fort Calhoun Station.

Item 3.a: Corrective Action Program

(1) Inspection Scope

An open item (Licensee Event Report), specifically related to component cooling water pump operations was reviewed by the team. The team verified the adequacy of the licensee's causal analysis and extent of condition evaluation. In addition, the team verified that adequate corrective actions were identified associated with the licensee's root and contributing causes and extent of condition evaluations, and that, implementation of these corrective actions are either implemented or appropriately scheduled for implementation.

(2) Observations and Findings

The team reviewed Licensee Event Report 2012-006, "Operation of Component Cooling Pumps Outside of the Manufacturers Recommendation," dated June 25, 2012. During this review, the team noted that during additional investigations conducted by the licensee, it had been determined that the flow instrumentation used during the testing was inaccurate and this caused invalid data to be used when assessing pump performance. Based on this, the licensee determined that the pumps had been operated as designed and not outside of manufacturer's recommendations. The licensee retracted LER 2012-006 via letter LIC-12-182, "Withdrawal of Licensee Event Report 2012-006, Revision 0, for the Fort Calhoun Station," dated December 12, 2012.

(3) Assessment Results

The team reviewed the licensee's testing data as well as the subsequent investigation data and determined that the licensee's conclusion to retract Licensee Event Report 2012-006, "Operation of Component Cooling Pumps Outside of the Manufacturers Recommendation," was appropriate.

This restart checklist item is closed.

Item 3.b: Equipment Design Qualifications

(1) Inspection Scope

a. Open items specifically related to maintaining systems, structures, and components within their licensing and design basis were reviewed by the team. Specifically, the team reviewed Restart Checklist Item 4.6.1.3 to assess the licensee's actions related to deficiencies that had been identified in the steam generator accident ring analyses. The inspection verified that the licensee resolved the deficiencies in the structural calculations by including the potential accident loads on major subcomponents of the steam generators. The team also reviewed an independent sample of other reactor coolant system structural calculations.

The team verified that the licensee performed adequate causal and extent of condition evaluations and that corrective actions are either implemented or appropriately scheduled for implementation.

b. Open items (Licensee Event Reports) related to pump mechanical seals and unanalyzed welds in the reactor coolant system were reviewed by the team. The team verified the adequacy of the licensee's causal analyses and extent of condition evaluations. In addition, the team verified that adequate corrective actions were identified associated with the licensee's root and contributing causes and extent of condition evaluations, and that, implementation of these corrective actions are either implemented or appropriately scheduled for implementation.

(2) Observations and Findings

a. <u>CAL Action Item 4.6.1.3 (Provide analysis of Steam Generator accident ring)</u>

The team's review of the selected calculations identified several significant errors with the calculations and inadequate extent of condition reviews. The apparent cause analysis report generated for Action Item 4.6.1.3 was narrowly focused. The licensee failed to analyze significant loads for a large component on the steam generator. The licensee's apparent cause stated, "Intimate knowledge of the effort led to complacency during the review, and the omission was not identified." The report focused on communication issues that occurred between various vendors, suppliers, and the licensee. Despite the fact that structural supports were removed during the steam generator replacement project, and loads were increased, a major structural component was not analyzed for design loads. Further, the extent of condition review determined there were no other errors or omissions in all calculations supporting the replacement steam generator design report. During the NRC inspection the team uncovered a number of errors that were not identified by the licensee's reviews.

The team noted that details in the calculations were challenging to follow. The licensee did not originate the calculations; an outside contractor prepared them. The licensee's staff was unable to effectively discuss the calculations with the team involving the calculation methodology, license basis requirements, and conclusions, without the vendor who originated them.

The team determined that the licensee had failed to provide adequate oversight over the contractor's preparation of the replacement steam generator calculation because the vendor utilized several inputs in the analyses that were not in conformance with the station's licensing basis. Further, the team found an example in the reactor coolant system structural calculations where the licensee had derived allowable stresses from vendor manuals, but did not actually possess the vendor manual. The licensee generated CRs 2013-14540 and 2013-14741 in response to this concern, and ultimately procured the vendor manual. The overarching issue of vendor manuals and vendor oversight was previously discussed in NRC Inspection Report 05000285/2013-008 (Accession No. ML13197A261), and was on the Restart Checklist as Item 3.d.1.

The team noted that the engineering staff continues to demonstrate gaps in their knowledge and understanding of the station's design basis with respect to load combinations. A specific example of this occurred during interviews related to the structural adequacy of the reactor coolant system. Specifically, the team questioned why it was acceptable for stress ratios to exceed the code allowable stress limits for a maximum hypothetical earthquake in conjunction with a maximum accident load (typically a loss of coolant accident). Station personnel generated CR 2013-14211 and an operability evaluation to address the team's concerns. The inspectors noted that the licensee's basis for the immediate operability determination stated, in part that, "the stress of the node occurs with a Maximum Hypothetical Earthquake and a design basis LOCA concurrently … this load combination is beyond design basis for the plant." The team determined that this was contrary to the facility current licensing basis because this combination is specifically addressed in the Updated Safety Analysis Report and other design and licensing basis documents. The licensee agreed that was a design basis load combination and generated CR 2013-19956 to capture this issue in the station's corrective action program.

The team also determined that the licensee was using a non-conservative procedure in the design of safety-related structures, systems, and components, and for evaluating degraded conditions. Specifically, the team noted that criteria from Station Procedure PED-MEI-17, "Interim Operability Criteria," (IOC) was inappropriately developed and applied to critical quality equipment (CQE) and and limited critical quality equipment (L-CQE) piping and pipe supports. The team determined that PED-MEI-17 had been inappropriately used, in some cases, by the engineering department to bypass evaluating non-conforming components using the operability process and entering the non-conformances into the corrective action program for timely resolution. In addition, the team noted that the licensee had made a commitment to notify the NRC each time they invoked the IOC procedure, but at some point in the past, the station failed to make required notifications.

During discussions with the licensee, the team was informed that the IOC operability limits contained in PED-MEI-17 were developed based on another licensee's IOC procedure, and the other licensee had received a safety evaluation report for use of IOC. The team requested a copy of the other licensee's IOC criteria and the safety evaluation report associated with it.

Subsequently, the team determined that the other licensee did not have a safety evaluation report for their IOC. Additionally, the team determined that the IOC limits contained in PED-MEI-17 were significantly less conservative than the other licensee's IOC limits from which they were supposedly based. The other licensee's IOC operability limits mirrored the faulted allowable stresses permitted by ASME Section III, Appendix F. ASME Section III, Appendix F, is generally endorsed by the NRC in Inspection Manual Chapter 0326, and by performing a comparison of the allowable stresses from ASME and PED-MEI-17, the team determined that: (1) the PED-MEI-17 operability limits were significantly less conservative than the ASME code allowable limits; (2) PED-MEI-17 did not contain all of the restrictions required by Appendix F. Therefore, the team determined that the IOC operability criteria was

non-conservative, and therefore, not suitable for operability determinations and not appropriate for use in design calculations.

As a result of the team's concerns with the use of IOC the licensee performed a review of corrective action reports and calculations to identify where the IOC was applied. In addition, as an immediate corrective action, the station discontinued the use of the IOC procedure at the station.

The team identified the following deficiencies during their review:

- NCV 05000285/2013013-06, "Failure to control deviations from the design basis requirements for structural calculations related to the reactor coolant system"
- NCV 05000285/2013013-07, "Programmatic Failure to Evaluate Safety Impact of Degraded Conditions during use of Interim Operability Criteria"
- NCV 05000285/2013013-08, "Failure to Correct Overstressed Components"
- NCV 05000285/2013013-09, "Non-conservative criteria in operability procedure"

These issues are further discussed in Section 5 of this report.

b. The team reviewed the following open items:

LER 2013-006	Low Pressure Safety Injection and Containment Spray Pumps Mechanical Seals
LER 2012-016	Unanalyzed Charging System Socket Welds to the Reactor Coolant System

The team verified the adequacy of the licensee's causal analyses and extent of condition evaluations. In addition, the team verified that adequate corrective actions were identified associated with the licensee's root and contributing causes and extent of condition evaluations, and that, implementation of these corrective actions are either implemented or appropriately scheduled for implementation.

(3) Assessment Results

a. The team concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews, and corrective actions taken or planned to be implemented, that the licensee has adequately addressed Restart Checklist Item 4.6.1.3.

Restart Checklist Item 4.6.1.3 is closed.

b. The team concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews, and corrective actions taken or planned to be implemented, that the licensee has adequately addressed the following LER's:

LER 2013-006	Low Pressure Safety Injection and Containment Spray Pumps Mechanical Seals
LER 2012-016	Unanalyzed Charging System Socket Welds to the Reactor Coolant System

With respect to LER 2013-006, "Low Pressure Safety Injection and Containment Spray Pumps Mechanical Seals" the licensee identified that the pump mechanical seals were made of a Teflon material that may not maintain the integrity of the system under accident conditions. The licensee corrected this deficiency by replacing the affected mechanical seals with seals qualified for the environmental conditions they would be subject to under design basis accident conditions.

With respect to LER 2012-016, "Unanalyzed Charging System Socket Welds to the Reactor Coolant System," the licensee identified that the chemical volume and control system (CVCS) inappropriately used socket welded fittings and the piping was in an unanalyzed condition involving thermal cycle fatigue. The licensee corrected these deficiencies by replacing affected piping and completing the thermal fatigue calculations for all affected piping.

These two LER's and associated Restart Checklist Items are closed.

Item 3.c.2: 10 CFR 50.59 Screening and Safety Evaluations

(1) Inspection Scope

After inspection of the licensee's program and conduct of 10 CFR 50.59 Screening and Safety Evaluations, which was documented in NRC Inspection Report 05000285/2013008, Restart Checklist Item 3.c.2, "10 CFR 50.59 Screening and Safety Evaluations," remained open. The decision by the team to leave the area open was based on the team's inability to close Restart Checklist Bases Document Items 3.c.2.2, "Adequacy of extent of condition and extent of causes," and 3.c.2.3, "Adequacy of corrective actions," for the root cause analysis for the 10 CFR 50.59 process.

The team reviewed licensee actions taken to address this area. For this follow-up review of the licensee's 10 CFR 50.59 process the team evaluated the thoroughness of their extent of condition and causal analysis, and the adequacy of identified corrective actions to ensure proper treatment of changes to the facility.

(2) Observations and Findings

<u>Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem</u>

During a previous Inspection Manual Chapter 0350 Confirmatory Action Letter Inspection documented in NRC Inspection Report 05000285/2013008, the team determined that the licensee's root cause evaluation did not fully address the extent of condition and the extent of cause of the problem. The team determined that the scope of the licensee's root cause analysis focused on events within the past five years for the extent of condition and the extent of cause of the problem. However, a number of plant changes were identified by that inspection team outside the scope of the 50.59 root cause analysis review period that failed to receive prior NRC review and approval before implementation.

To address this observation, the licensee expanded their scope. The licensee first expanded scope of their 10 CFR 50.59 reviews back to the year 2005. A subsequent expansion back to the year 2000 was conducted as a result of the review of their root cause analysis. Additionally, as a long term corrective action the licensee has committed to implement a design basis reconstitution project that addresses ensuring system design requirements are established for all safety significant systems. Based on these actions the NRC determined the licensee is adequately addressing the extent of condition and extent of cause of the problem.

<u>Determine that appropriate corrective actions are specified for each root and contributing cause</u>

During a previous Manual Chapter 0350 Confirmatory Action Letter Inspection documented in NRC Inspection Report 05000285/2013008, the team determined that the licensee specified appropriate corrective actions for each root and contributing cause. However, the team identified that all corrective actions to prevent reoccurrence for the root causes were not in place and effective.

Specifically, one corrective action by the licensee implemented a team to evaluate all engineering changes as an interim action. The licensee called the team, established in accordance with this corrective action, the Engineering Assurance Group (EAG). The team questioned the effectiveness of the EAG relative to 10 CFR 50.59 evaluations after discovering that the group had reviewed an evaluation for the station's tornado missile design and came to a different conclusion than the NRC on the need for a license amendment.

Also, the Manual Chapter 0350 Confirmatory Action Letter inspection team determined that actions taken had not fully addressed the need for the station to update their current licensing basis documents and for the licensee to train the Fort Calhoun Station personnel to understand those documents. The team concluded that changes to the facility would be impacted by the incomplete understanding of the existing design and licensing bases.

To address these observations, the licensee conducted additional training for the EAG on the 10 CFR 50.59 program. After this, the team observed that a subsequent major design change for high energy line break analysis was properly evaluated by the licensee per 10 CFR 50.59. The licensee also developed tracking metrics to monitor the health of the 10 CFR 50.59 program at the station. Finally, the licensee committed to a long term project to review and update the design and licensing basis of the station.

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

During the previous Manual Chapter 0350 Confirmatory Action Letter Inspection, documented in NRC Inspection Report 05000285/2013008, the team determined that the licensee established a schedule for implementing and completing some of the corrective actions, and that, one key action had not been completed. The licensee had scheduled the initial training for March 15, 2013. However, the licensee had moved the training to an undetermined date. At that time, the team concluded that the failure of the licensee to not establish or assign a new date was insufficient to consider this aspect as resolved.

To address this observation, the licensee completed 10 CFR 50.59 training classes for both evaluators as well as screeners, which were specifically targeted to past noted deficiencies. The initial round of this training was completed in April 2013. Another session of this course for additional personnel was planned.

(3) Assessment Results

After reviewing actions taken for gaps noted in the licensee's 10 CFR 50.59 program and process, documented in NRC Inspection Report 05000285/2013008, the team concluded that the licensee had adequately addressed their deficiencies relative to the 10 CFR 50.59 program.

The following Restart Checklist Items for Area 3.c are closed:

3.c.2.2 Adequacy of extent-of-condition and extent of causes

3.c.2.3 Adequacy of corrective actions

Item 3.d: Maintenance Programs

(1) Inspection Scope

The team reviewed the licensee's assessment of the Fundamental Performance Deficiency associated with equipment reliability and work management. Specifically, the team assessed CR 2012-8134, for which the "Description" section stated, in part:

> *"Equipment problems are not prevented, identified, or resolved in a thorough and timely manner. Issues contributing to this problem include intolerance to*

equipment failures has not been established, long term strategies have not been developed for age related degradation, the maintenance rule function to monitor the performance of plant equipment has not been effectively implemented, and work activities are not effectively managed to ensure long-term equipment reliability. As a result, the station has experienced low levels of equipment reliability that affect nuclear safety and work management practices challenge the safe and reliable operation of the plant."

The team also assessed the adequacy of the extent of condition, extent of causes, and corrective actions.

The team's assessment of this Fundamental Performance Deficiency was based on the evaluation criteria from Section 02.02 of NRC Inspection Procedure 95001, which aligns with this item. The inspection objectives were to:

- Provide assurance that the root and contributing causes of risk-significant issues were understood;
- Provide assurance that the extent-of-condition and extent-of-cause of risk-significant issues were identified; and
- Provide assurance that the licensee's corrective actions for risk-significant performance issues were, or will be, sufficient to address the root and contributing causes and preclude repetition.

(2) Observations and Findings

<u>Determine that the problem was evaluated using a systematic methodology to identify the root and contributing causes</u>

The team determined that the licensee evaluated this problem using a systematic methodology to identify the potential root and contributing causes. Specifically, Root Cause Analysis 2012-08134 used the analytical techniques of event and causal factor charting and barrier analysis to identify causal relationships. A safety culture evaluation was also completed as part of the analytical process.

The licensee identified the following as the root cause and contributing causes:

RC-1: Fort Calhoun Station senior leadership failed to ensure corrective actions were taken to address safety issues, adverse trends, and assessment-revealed issues that were identified in the Equipment Reliability programs and processes.

CC-1: Management has not applied an industry-standard Plant Health Committee process to ensure the success of Equipment Reliability programs and processes.

CC-2: The training programs or qualification processes have not been fully effective to ensure station personnel have satisfactory skills and knowledge

enabling them to execute needed work management and long-term equipment reliability functions.

CC-3: The station leadership team has not demonstrated accountability nor held station personnel accountable for implementation of the engineering and work management processes in support of long-term equipment reliability.

CC-4: Procedure and process deficiencies have contributed to the degraded equipment reliability issue.

CC-5: Fort Calhoun Station failed to ensure that equipment reliability programs, including regulatory required Maintenance Rule program and the supporting PM program, were adequately staffed, funded, and trained, resulting in the inability to identify, correct, and prioritize equipment problems which resulted in the unacceptable performance of certain safety related structures, systems, and components.

<u>Determine that the root cause evaluation was conducted to a level of detail commensurate with the significance of the problem</u>

The licensee conducted the evaluation to a level of detail commensurate with the significance of the problem. The root cause team interviewed various levels of site personnel and evaluated station procedures, documents, condition reports, internal/external operating experience, and related contractor reports.

<u>Determine that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience</u>

The licensee reviewed internal and external operating experience to determine whether the same of similar problems have previously occurred at the Fort Calhoun Station or within the industry, and if so, what lessons can be learned for the Fort Calhoun Station. The review also determines if the Problem Statement falls within the definition for a ‘Repeat Event’.

The licensee determined the use of operating experience was not implicated as a cause/contributor to the condition investigated by this Root Cause Analysis.

<u>Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem</u>

The licensee determined that the conditions discussed in the root cause analysis continue to impact the reliability of plant structures, systems, and components. Corrective actions to address the conditions are not short term and require the restoration, and in some cases, the rebuilding of the programs that have been allowed to decay over the past few years. In addition, while the Maintenance Rule and the preventative maintenance (PM) programs are the primary programs that affect the equipment issues raised by this condition report, there are many more focused programs that support these programs, such as the Motor Operated Valve

program, the Air Operated Valve program, Flow Accelerated Corrosion program, and others. All of these would be affected by the cause of this issue since management's failure to understand the requirements of an effective reliability effort would extend to any program that dealt with equipment reliability.

The licensee has determined that an extent of condition exists.

The licensee evaluated the potential extent of cause for Root Cause 1. The licensee determined this cause extended to engineering issues, and procedural issues that were identified as part of this investigation. There were multiple instances where conditions/issues were identified internally or externally, identified repetitively, but never fixed. When an issue was identified, the Fort Calhoun Station wrote a condition report, instituted a program (BOM, EROP), and then did not ensure that these actions addressed the identified shortcoming. There is an Extent of Cause as this issue applies to the entire Corrective Action Program, and thus, to the entire station.

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

The root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in Inspection Manual Chapter 0310. The safety culture review evaluated safety culture aspects against the data collected during the cause evaluation. Their review identified the cross-cutting aspects of P.1(d), P.3(c), and P.1(c), were the most applicable.

<u>Determine that appropriate corrective actions are specified for each root and contributing cause</u>

The team reviewed the licensee's corrective actions for each of the root and contributing causes. RC-1 is addressed by the corrective action to prevent recurrence, CAPR-1, AI 2012-03986-009, listed in the Organizational Ineffectiveness at the Fort Calhoun Station RCA. It addresses the oversight and accountability for Nuclear Safety at all of Fort Calhoun Station to include the cultural aspect of a Continuous Learning Environment. CAPR-2 revises Station Procedure FCSG-33, "FCS Issue prioritization and Plant Health Committee Process, to improve the processes of Plant Health Committee (PHC)."

<u>Determine that a schedule has been established for implementing and completing the corrective actions</u>

The team identified that within Root Cause Analysis 2013-08134 a schedule had been established for implementing and completing the assigned corrective actions. At the time of the inspection, the corrective actions to prevent recurrence had been completed and a few of the other corrective actions for the contributing causes had been designated as complete. The team noted that some of the important corrective actions related to the engineering program's issues, such as revising the Preventive Maintenance Program, were not due to be completed until 2014. The team felt that

these key engineering programs, gaps identified in the licensee's Equipment Reliability Restoration Plan, and coordination of system and component maintenance activities within the work management process, should have a higher priority so as to address these potentially significant conditions in a timelier manner.

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

The inspectors noted the licensee had not established specific criteria to assess the effectiveness of corrective actions to prevent recurrence. However, equipment issues would be documented in the condition reporting system and screened based on risk and safety significance for causes. The tracking and trending of these issues provides reasonable assurance the licensee should detect ineffective corrective actions.

(3) Assessment Results

The team concluded, based on their reviews of the licensee's cause evaluations and the extent of cause/extent of condition reviews, that this area has been adequately addressed by the licensee.

The following Restart Checklist Items are closed:

3.d.1 Licensee Assessment of the Fundamental Performance Deficiency associated with Equipment Reliability/Work Management

3.d.2 Adequacy of extent-of-condition and extent of causes

3.d.3 Adequacy of corrective actions

Item 3.d.2: Equipment Service Life

(1) Inspection Scope

a. The team reviewed the licensee's assessment of the engineering area associated with Equipment Service Life. Specifically, the team assessed CR 2012-9491, for which the "Problem Statement" section said, in part,

> *"FCS has operated some equipment beyond its service life."*

The team also assessed the adequacy of the extent of condition, extent of causes, and corrective actions.

The team's assessment of this area was based on the evaluation criteria from Section 02.02 of NRC Inspection Procedure 95001, which aligns with this item. The inspection objectives were to:

- Provide assurance that the root and contributing causes of risk-significant issues were understood;

- Provide assurance that the extent-of-condition and extent-of-cause of risk-significant issues were identified;

- Provide assurance that the licensee's corrective actions for risk-significant performance issues were, or will be, sufficient to address the root and contributing causes and to preclude repetition.

b. Restart Checklist Item NCV 2011003-04, "Failure to Provide Procedural Guidance to Replace or Evaluate Age Degraded Components," was reviewed by the team. The team verified the adequacy of the licensee's causal analysis and extent of condition evaluations related to this issue. In addition, the team verified that adequate corrective actions were identified and associated with the licensee's root and contributing causes and extent of condition evaluations, and that, implementation of these corrective actions are either implemented or appropriately scheduled for implementation.

(2) Observations and Findings

a. <u>Licensee's Evaluation of Equipment Service Life Issues</u>

<u>Determine that the problem was evaluated using a systematic methodology to identify the root and contributing causes</u>

The team determined that the licensee evaluated this problem using a systematic methodology to identify the potential root and contributing causes. Specifically, Root Cause Analysis 2012-9491 used the analytical techniques of event and causal factor charting, process fault tree, common factors chart, and barrier analysis to identify causal relationships. A safety culture evaluation was also completed as part of the analytical process.

The licensee identified the following as the root cause and contributing causes:

RC-1: Leadership failed to provide the level of command and control needed to prevent Preventative Maintenance (PM) programmatic weaknesses. Shortfalls include inaccurate or incomplete procedures and programmatic documents, incomplete PM bases, inconsistent use of end of service life (EOSL) tools, inadequate system monitoring, and insufficient replacement strategies for components beyond EOSL. This resulted in the design and implementation of the station's preventative maintenance (PM) program to not meet industry standards for operating components beyond end of service life.

CC#1: PM program improvements since 2005 were not effectively managed resulting in ongoing programmatic deficiencies. For example, resources were not managed to ensure Equipment Reliability Optimization Project (EROP) PMs

were developed and implemented, oversight did not ensure components were correctly scoped, and project plans did not identify equipment at EOSL.

CC#2: Corrective action program behaviors to resolve PM programmatic weaknesses that would have addressed component EOSL activities were ineffective. Deficiencies were identified multiple times since 2005.

<u>Determine that the root cause evaluation was conducted to a level of detail commensurate with the significance of the problem</u>

The licensee conducted the evaluation to a level of detail commensurate with the significance of the problem. The root cause team interviewed various levels of site personnel and evaluated station procedures, documents, condition reports, internal/external operating experience, and related contractor reports.

<u>Determine that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience</u>

The licensee reviewed internal and external operating experience to determine whether the same of similar problems have previously occurred at the Fort Calhoun Station or within the industry, and if so, what lessons can be learned for Fort Calhoun Station. The review also determines if the Problem Statement falls within the definition for a 'Repeat Event'.

The licensee determined that in many situations, the station had opportunities to identify the overall problems with equipment service life, but tended to focus only on the issues included in the condition reports. The plant developed corrective actions to address the specific conditions being evaluated, but did not address the larger issues.

The licensee determined the use of operating experience was not implicated as a cause/contributor to the condition investigated by this Root Cause Analysis.

<u>Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem</u>

The licensee evaluated the potential extent of condition that noncritical equipment may have been operated beyond its service life. They also evaluated whether other programs governing operation of equipment required for safe and reliable operation of the station may have deficiencies that result in critical equipment operating in an unreliable condition. The potential extent of condition is the incomplete status of station programs intended to improve equipment reliability, including the following:

- PM Program Basis
- System / Component Performance Monitoring
- Life Cycle Management
- Functional Importance Determination
- Component Obsolescence Program

- Bill of Materials Development Project
- PM Work Order Task Upgrade Project
- EROP/First Time PMs

The licensee has determined that an extent of condition exists.

The licensee evaluated the potential extent of cause for Root Cause 1. The licensee determined that an extent of cause exists.

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

The root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in IMC 0310. The safety culture review evaluated safety culture aspects against the data collected during the cause evaluation. Their review identified the cross-cutting aspects of H.2(a), H.2(c), P.1(c), and O.2(b) as the most applicable.

<u>Determine that appropriate corrective actions are specified for each root and contributing cause</u>

The team reviewed the licensee's corrective action for each of the root and contributing causes. The corrective actions to prevent recurrence were to: (1) revise or replace FCSG-33, "FCS Issue Prioritization and Plant Health Committee Process," and; (2) improve the processes of the Plant Health Committee and develop and implement a PM program with component EOSL strategy that meets the industry standards.

<u>Determine that a schedule has been established for implementing and completing the corrective actions</u>

Due dates are established for corrective actions for CR 2012-9491. At the time of the inspection, corrective action to prevent recurrence 1 had been completed and a few of the other corrective actions for the contributing causes had been completed. The team noted that the corrective action to prevent recurrence 2, which addresses the service life documentation issue, is not due until March 31, 2014. The licensee has evaluated all safety related components to determine actions necessary prior to returning the unit to service.

During their review the team determined that the licensee had failed to provide an adequate basis for operability for components that were identified as being past their specified service life. The team identified this performance deficiency as, NCV 05000285/2013013-09, "Failure to Follow Operability Procedure." This issue is further discussed in Section 5 of this report.

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

The inspectors noted the licensee had not established specific criteria to assess the effectiveness of corrective actions to prevent recurrence. However, equipment service life issues would be documented in the condition reporting system and screened based on risk and safety significance for causes. The tracking and trending of these issues provides reasonable assurance the licensee should detect ineffective corrective actions. Additionally, the licensee has long term actions to perform self-assessments of the equipment reliability, preventative maintenance and performance monitoring programs, including the Plant Health Committee oversight of equipment reliability.

b. The team reviewed the licensee’s causal analyses, corrective actions, and extent of condition associated with previously identified issue, NCV 05000285/2011003-04, “Failure to Provide Procedural Guidance to Replace or Evaluate Age Degraded Components.” The team verified that adequate corrective actions were identified associated with the causes and extent of condition evaluations and that these corrective actions were either implemented or appropriately scheduled for implementation.

(3) Assessment Results

a. The team has concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews, that this area has been reviewed by the licensee to a sufficient level of detail. The following Restart Checklist Items are closed:

3.d.2.1 Licensee Assessment of equipment service life program

3.d.2.2 Adequacy of extent-of-condition and extent of causes

3.d.2.3 Adequacy of corrective actions

3.4.1.1 Replace Non-RPS CQE (reactor protection system critical quality equipment) power supplies that will be beyond their recommended service life.

3.4.2.2 Identify all CQE power supplies; priority will be on RPS CQE power supplies and then non-RPS CQE power supplies.

3.4.2.3 Determine the installation date for FCS CQE power supplies; these dates will be used to define those CQE power supplies that are beyond their service life.

3.4.2.4 Conduct an industry and FCS specific analysis of historical performance for CQE power supplies; determine the effectiveness of the current Equipment Reliability (ER) Strategies at the FCS component level.

3.4.2.5 Conduct an analysis of the current FCS ER Strategy for power supplies; contact vendors, review industry documentation, and benchmark other plants.

3.4.2.6 Determine the recommended service life for CQE power supplies based on analyses performed earlier in this action plan.

These service lives will be based on: (1) manufacturer and model, (2) qualified life testing, (3) vendor recommendations and communication with vendors, (4) remnant life based on stress testing of removed power supplies, (5) industry and FCS specific historical performance, and (6) actual duty cycle and service condition where these power supplies are installed .

3.4.2.7 Conduct a failure modes and effects analysis on each power supply to ensure the impact of failures is understood.

3.4.2.8 Document the time based replacement strategy and basis for CQE and RPS power supplies. This strategy and basis will provide the tasks to be performed and the basis for the scope and frequency of those tasks. This action is being completed before start up to ensure each power supply has been analyzed and a recommended service life defined.

3.4.2.9 Define those power supplies that are beyond their service life. This will include power supplies that will be beyond their service life before the next planned refueling outage.

3.4.2.10 Replace RPS CQE power supplies beyond their service life.

3.4.2.11 Replace Non-RPS CQE power supplies that will be beyond their recommended service life.

b. The team concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews associated with the licensee’s response to NCV 05000285/2011003-04, “Failure to Provide Procedural Guidance to Replace or Evaluate Age Degraded Components,” that this item is closed.

4. Assessment of NRC Inspection Procedure 95003 Key Attributes

Section 5 of the Restart Checklist is provided to assess the key attributes of NRC Inspection Procedure 95003. The key attributes are listed as separate subsections below. It is intended that the activities in these subsections be conducted in conjunction with reviews and inspections for Sections 1 – 4, rather than a stand-alone review. In addition, the NRC will review the effectiveness of licensee short term and long term corrective actions associated with these areas to ensure they are adequate to support sustained plant performance improvement.

Item 5.a: Design

(1) Inspection Scope

a. The team independently assessed the extent of risk significant design issues. The review covered the as-built design features of the auxiliary feedwater system. This review verified its capability to perform its intended functions with a sufficient margin of safety. The basis for selecting the auxiliary feedwater system was its high risk significance in the specific individual plant evaluation, and input from system health reports, performance indicators, condition reports, and licensee event reports. Focus

was on modifications rather than original system design. Information from this inspection was used to assess the licensee's ability to maintain and operate the facility in accordance with the design basis.

The team's review included the following:

- Assessment of effectiveness of corrective actions for deficiencies involving design

- Selection of several modifications to the auxiliary feedwater system to determine if the system is capable of functioning—as specified by the current design and licensing documents, regulatory requirements, and commitments for the facility

- Determination if the auxiliary feedwater system is operated consistent with the design and licensing documents

- Evaluation of the interfaces between engineering, plant operations, maintenance, and plant support groups

b. The team reviewed the licensee's assessment of the Fundamental Performance Deficiency associated with Engineering Design/Configuration Control. Specifically, the team assessed the RCA associated with CR 2012-08125, for which the problem statement was:

> "Changes to plant configuration and design and licensing bases are not effectively analyzed, controlled, and implemented. These change processes are not always conducted in a manner that maintains configuration control and operating design margins."

The team also assessed the adequacy of the extent of condition, extent of causes, and corrective actions.

The team's assessment of this Fundamental Performance Deficiency was based on the evaluation criteria from Section 02.02 of NRC Inspection Procedure 95001 which aligns with this item. The inspection objectives were to:

- Provide assurance that the root and contributing causes of risk-significant issues were understood;

- Provide assurance that the extent-of-condition and extent-of-cause of risk-significant issues were identified;

- Provide assurance that the licensee's corrective actions for risk-significant performance issues were, or will be, sufficient to address the root and contributing causes and to preclude repetition

c. Restart Checklist Item NCV 2010006-01 specifically related to the failure to correct repeated tripping of the turbine driven auxiliary feed water pump was reviewed by the team. The team verified the adequacy of the licensee's causal analysis and extent of condition evaluations related to and associated with the issue. In addition, the team verified that adequate corrective actions were identified and associated with the licensee's root and contributing causes and extent of condition evaluations, and that, implementation of these corrective actions are either implemented or appropriately scheduled for implementation.

(2) Observations and Findings

a. <u>Auxiliary Feedwater System Design Review</u>

The team completed an in depth assessment of select risk significant design issues associated with the auxiliary feedwater system. During this review the team identified some issues associated with the auxiliary feedwater system. Specifically:

- NCV 05000285/2013013-10, "Failure to Evaluate the Effects of Modifying the Turbine Driven Auxiliary Feedwater Pump"
- NCV 05000285/2013013-16, "Failure to Submit Licensee Event Report" (Example 3)

These specific issues are documented in Section 5 of this report.

b. <u>Fundamental Performance Deficiency Review</u> <u>Deficiency Associated with Engineering Design/Configuration Control</u>

<u>Determine that the problem was evaluated using a systematic methodology to identify the root and contributing causes</u>

The team determined that the licensee evaluated this problem using a systematic methodology. Specifically, the licensee developed comparative timelines, a common factors chart, and conducted a barrier analysis to complete Root Cause Analysis 2013-05570, "Design and Licensing Bases Configuration Control." However, the licensee did not strictly follow the process in all cases for using the systematic reviews to identify the root and contributing causes. Specifically, Root Cause Analysis 2013-05570 documented the following root causes:

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.

The team noted that RC-1 more closely fits the definition of a contributing cause in station procedures. For instance, to supplement RC-1 the licensee stated: "This

initial condition [the incomplete design and licensing bases], combined with a weakness in licensing bases knowledge and a failure to internalize the importance of the design bases, resulted in the organization missing repeated opportunities to correct the initial deficiencies and additional errors were created over time." A root cause is defined in Station Procedure FCSG-24-4, Attachment 1, Section 1.17, as the most basic, fundamental cause(s) of a problem, which, if corrected, will prevent recurrence of the identified problem and similar problems. When evaluated against the cause testing criteria used by the licensee and described in Station Procedure FCSG-24-5, "Cause Evaluation Manual," the team concluded that RC-1, without accounting for the knowledge aspect, does not, by itself, constitute a root cause.

Similarly, when applying the cause test to RC-2, the team concluded that the following cause test questions could have been answered "Yes", suggesting that RC-2 is a contributing and not a root cause: (1) If this cause being considered was absent, would the event that initiated the evaluation have occurred?; (2) If this cause is eliminated, is there a way for the same event to occur?; and (3) If this cause is eliminated, will there be future similar events?

<u>Determine that the root cause evaluation was conducted to a level of detail commensurate with the significance of the problem</u>

The team determined that the root cause analysis was conducted to a level of detail commensurate with the significance of the problem. Specifically, as discussed above, the licensee conducted Root Cause Analysis 2013-05570 using comparative timelines, a common factors chart, and a barrier analysis. The analysis was also supplemented by information gathered through interviews and a historical overview which helped illustrate the magnitude and precedence of Fort Calhoun Station's inability to maintain design control and documentation associated with structures, systems, components, and activities affecting quality. The licensee's root cause analysis techniques were generally thorough and to a level of detail commensurate with the significance of the problem.

<u>Determine that the root cause evaluation included a consideration of prior occurrences of the problem and knowledge of prior operating experience</u>

The team determined that the root cause analysis included evaluations of both internal and external industry operating experience. The licensee's evaluations of industry operating experience provided sufficient detail such that general conclusions could be established regarding any similarities. The root cause analysis team's operating experience review also determined this problem fell within the definition of a repeat event. In accordance with Station Procedure FCSG-24-4, "Condition Report and Cause Evaluation," a repeat event is a significance Level A condition or event that shares the same or similar root causes as a previous event. The root cause analysis write up stated that, while the team did not identify similar corrective actions to prevent recurrence associated with a root cause, it is clear by a review of the timeline presented in the report that this event was preventable through the use of internal and external operating experience.

The team identified, however, that the licensee did not document the repeat event in accordance with station procedures. Specifically, Station Procedure FCSG-24-4 states that, if the problem is determined to be a repeat event then the root cause analysis shall explain why previous root cause analysis corrective actions to prevent recurrence did not prevent the repeat event, and the new corrective action to prevent recurrence should consider why the previous corrective actions to prevent recurrence were not effective. In addition, a condition report should be issued describing the problem with the previous root cause analysis(s) and reference the condition report in this section of the root cause analysis report. Although documented as a repeat event, the licensee did not perform the required actions. The specific issue is documented as NCV 05000285/2013013-11, "Failure to Perform Adequate Operating Experience Reviews In Accordance with Station Procedure FCSG-24-4." This issue is further discussed in Section 5 of this report.

<u>Determine that the root cause evaluation addressed the extent of condition and the extent of cause of the problem</u>

The team reviewed Root Cause Analysis 2013-05570 as it relates to extent of condition and extent of cause.

For the extent of condition, the licensee evaluated the extent to which the actual condition existed with other plant processes, equipment, or human performance. The condition, in this case, is that the licensee did not maintain adequate configuration control of the structures, systems, components, or activities in accordance with 10 CFR Part 50, Appendix B. The licensee used the approaches of Station Procedure FCSG-24--4, "Course Evaluation Manual," for their review and concluded that there was no extent of condition. In their review, the licensee stated that the problem includes all station structures, systems, components, and processes encompassed by the design and licensing bases, and as such, it could not cause further impact to other structures, systems, components, or processes. The team noted that overall, the licensee's extent of condition review was superficial and the answers were broad. Essentially, the licensee presumed that because the problem statement is so broad, it implicitly includes every plant process that is impacted by the problem. Consequently, the licensee saw no need to specifically list them in the review. However, the team noted that since other processes are significantly impacted by this problem, listing them as part of the review would have generated corrective actions associated with each specific process. For instance, processes such as operability determination, 50.59 reviews, configuration control (tagging), design, vendor modifications, work control, surveillance program, preventive maintenance, and nondestructive examination would be impacted by the licensee's failure to maintain adequate configuration control of the structures, systems, components, or activities, in accordance with, 10 CFR Part 50, Appendix B.

For the extent of cause, the licensee reviewed the root causes of the identified problems to determine where they may have impacted other plant processes, equipment, or human performance. The licensee concluded that RC-1 extended to

the procedures of other site organizations that could have been incorrectly translated or impacted due to the lack of knowledge and understanding of design and licensing bases. Specifically, the licensee considered the following departments and processes as being impacted: Radiation Protection, Emergency Planning, Chemistry, Security, Operations procedures, Maintenance procedures, and Engineering implementing procedures. The team noted that the extent of cause did not document the basis for the vulnerable/not vulnerable conclusion for each area of the potentially vulnerable list as required in Station Procedure FCSG-24-5.

<u>Determine that the root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in Inspection Manual Chapter 0310</u>

The root cause, extent of condition, and extent of cause evaluations appropriately considered the safety culture components as described in Inspection Manual Chapter 0310. The licensee identified that a majority of the cross-cutting aspects were applicable to issues related to the station's inability to maintain design control and documentation associated with structures, systems, components, and activities affecting quality. Specifically, the areas of human performance, problem identification and resolution, safety conscious work environment, and other components were applicable to issues related to design and licensing bases maintenance.

<u>Determine that appropriate corrective actions are specified for each root and contributing cause</u>

The team reviewed the licensee's corrective actions for each of the root and contributing causes for both root cause analyses. The corrective actions to prevent recurrence and implemented to address the root causes identified in Root Cause Analysis 2013-05570, were to identify and define the licensing and design bases and assure licensing and design bases documentation remains current, accurate, complete, and retrievable. The corrective action to prevent recurrence also included modifying the engineering support personnel initial and continuing training programs to incorporate the corrective action to prevent recurrence previously mentioned (the identification and definition of licensing and design bases to assure they remain current, accurate, complete, and retrievable). Lastly, the licensee stated that, as an additional corrective action to prevent recurrence, they would strengthen the function of the oversight group that performs reviews of documentation, including 10 CFR 50.59 reviews, modifications, operability evaluations, and other documents developed that utilized design and licensing bases information. Other corrective actions included: (1) providing training to personnel who utilize the design and licensing bases, including the individuals involved with the processes already mentioned; (2) developing and implementing performance metrics for the implementation of the corrective action to prevent recurrence and corrective actions mentioned.

The team determined that the corrective actions identified for the root and contributing causes appear to be adequate in principle. However, the team noted

that the due dates for the corrective actions to prevent recurrence are set in the distant future, and as a result, it will be a significantly long time before all the actions to address the licensee's inability to maintain design control and documentation associated with structures, systems, components, and activities affecting quality, will prevent recurrence of these issues. At the time of this inspection, none of the corrective actions to prevent recurrence or corrective actions associated with this root cause analysis had been completed.

The team also reviewed several interim actions implemented by the licensee. Interim actions were taken to temporarily prevent the effects of a condition or make an event less likely to recur during the period when final corrective actions or corrective actions were completed. The team noted that, as part of the interim action, the licensee completed an operability evaluation to allow the use of the Alternate Seismic Criteria Methodology (ASCM) to support plant startup. However, the NRC had already communicated with the licensee that the use of ASCM is not permissible.

The team identified the following deficiencies during their review:

- NCV 05000285/2013013-13, "Failure to Incorporate Design Requirements For Switchgear Room Cooling"

- NCV 5000285/2013013-14, "Inadequate Corrective Action for Non-Seismic Category 1 Piping"

- NCV 05000285/2013013-15, "Lack of an Adequate Operability Evaluation for Class 1 Raw Water Piping in Non-Class 1 Service Building"

- NCV 05000285/2013013-16, "Inadequate Operability Determination due to Failure to Consider an Unavailable Raw Water Pump"

- NCV 05000285/2013013-17, "Failure to Translate Design Sluice Gate Leakage Into Operating Procedure"

- NCV 05000285/2013013-18, "Inadequate Procedure for Intake Cell Level Control During a Flooding Event"

- NCV 05000285/2013013-19, "Failure to Translate Appendix R License Exemptions into the Plant's Fire Protection Program Design"

- NOV 05000285/2013013-20, "Failure to Provide Complete and Accurate Information to the NRC"

- NCV 5000285/2013013-21, "Failure to Perform Adequate Extent of Condition Reviews"

- URI 05000285/2013013-22, "Shutdown Cooling Piping and Pipe Supports Calculation Has Incorrect Acceptance Criteria for Anchor Displacement"

These issues are further discussed in Section 5 of this report.

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

The team determined that a schedule has been established for implementing and completing the corrective actions associated with Root Cause Analysis 2013-05570. However, the team also noted that most of the corrective actions are scheduled for completion in the future, and the team was not able to verify them by the end of the inspection period. In addition, even though the licensee has implemented interim corrective actions, the team still found many issues with the licensee's design and licensing bases maintenance. Notwithstanding, the team concluded that due to the extent and magnitude of the corrective actions, the schedule for the dates established for completion appeared to be reasonable.

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

The licensee developed effectiveness reviews to measure the progress and success of the corrective action to prevent recurrence for Root Cause Analysis 2013-05570. The licensee established effectiveness reviews that will include, in part, the determination of the reconstitution of the design and licensing bases was implemented properly and in a timely manner. In addition, the licensee will check if there have been any recurring instances of failure to maintain the licensing bases. Furthermore, the licensee established interim effectiveness reviews that consist of periodic assessments tracking the progress of the reconstitution of the licensing bases. The reviews will evaluate the implementation of the reconstitution and determine if the milestones are met and documentation is retrievable. In addition, the interim effectiveness reviews will evaluate the determination of the records after they are established and before the actions to reconstitute records begin. These interim effectiveness reviews will occur every eight months.

The team noted that the effectiveness reviews have been determined/decided conceptually. However, at the time of this inspection (and because it is so early in the process), the licensee had no details established as to what the specific methodology to conduct the effectiveness reviews will be. Specifically, the licensee has established the dates of the effectiveness reviews, which will be conducted throughout the reconstitution of the licensing and design basis documents. However, the action items in Root Cause Analysis 2013-05570 do not provide detail of the process/methodology. At the time of this inspection, none of the effectiveness reviews were ready for inspection since, as mentioned before, the due dates are in the future.

c. The team reviewed the licensee's causal analyses, corrective actions, and extent of conditions associated with the previously identified issue, NCV 05000285/2010006-01, "Failure to Correct Repeated Tripping of the Turbine-driven Auxiliary Feedwater Pump FW-10." In addition, the team verified that adequate corrective actions were identified and associated with the causes and

extent of condition evaluations, and that, these corrective actions were either implemented or appropriately scheduled for implementation.

(3) Assessment Results

a. The team concluded, based on their engineering inspection activities associated with the auxiliary feedwater system, their reviews of the cause evaluations, and the extent of cause/extent of condition reviews, that this area has been adequately addressed by the licensee. The following Restart Checklist Items are closed:

   5.a.1 Perform NRC design engineering team inspection of the Auxiliary Feedwater System

   5.a.2 Licensee Assessment of the Fundamental Performance Deficiency associated with Engineering/Configuration Control

   5.a.3 Adequacy of extent-of-condition and extent of causes

   5.a.4 Adequacy of corrective actions

b. The team concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews associated with the licensee's response to NCV 05000285/2010006-01, "Failure to Correct Repeated Tripping of the Turbine-driven Auxiliary Feedwater Pump FW-10," that this item is closed.

<u>Item 5.d: Equipment performance</u>

(1) Inspection Scope

Restart Checklist Item LER 2012-018 related to the containment air cooling units being operated outside of Technical Specification requirements was reviewed by the team. The team verified the adequacy of the licensee's causal analyses and extent of condition evaluations. In addition, the team verified that adequate corrective actions were identified and associated with the licensee's root and contributing causes and extent of condition evaluations, and that, implementation of these corrective actions are either implemented or appropriately scheduled for implementation.

(2) Observations and Findings

The team reviewed the licensee's causal analyses, corrective actions, and extent of condition associated with Licensee Event Report 2012-018, "Containment Air Cooling Units Operated Outside of Technical Specification during Cycle 26." In addition, the team verified that adequate corrective actions were identified associated with the causes and extent of condition evaluations and that these corrective actions were either implemented or appropriately scheduled for implementation.

(3) Assessment Results

The team has concluded, based on their reviews of the cause evaluations and the extent of cause/extent of condition reviews associated with Licensee Event Report 2012-018, "Containment Air Cooling Units Operated Outside of Technical Specification during Cycle 26," that this item is closed.

5. **Specific Issues Identified During This Inspection**

(1) Introduction. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the licensee's failure to promptly identify and correct a condition adverse to quality.

Description. On May 2, 2012, the licensee completed Root Cause Analysis 2011-06621 associated with 480 Vac circuit breaker 1B3A tripping open due to excessive current draw during the fire event in the 480 Vac 1B4A load center that occurred on June 7, 2011. The licensee determined that zone select interlock jumpers for the 1B3A Nuclear Logistics Incorporated/Square-D Masterpact circuit breaker was incorrectly installed during the replacement of the original General Electric AK-50 low voltage power circuit breaker in the 480 Vac 1B3A load center. With the jumpers incorrectly installed, the zone select interlock feature for circuit breaker 1B3A was not disabled. This configuration resulted in the breaker 1B3A tripping at the instantaneous overcurrent setpoint (immediately) when it sensed a fault, instead of tripping at the appropriate timed overcurrent setpoint, which would have allowed bus tie breaker BT-1B3A to open, and not result in the loss of load center 1B3A during the fire event. The licensee also identified that injection testing with the full function test kit bypassed the zone select interface feature, regardless of the configuration of the zone select interface jumpers installed at the breaker. Therefore, the testing that had been performed would not have identified the zone select interface jumper issues. The licensee initiated corrective action item CR 2011-06621-32 to perform current injection testing on all 480 Vac breakers without the use of a full function test kit to ensure that the zone select interface does not adversely impact breaker coordination. The licensee documented that this action as complete on January 15, 2013.

The team reviewed Root Cause Analysis 2011-06621, and its associated corrective actions. The team noted that 10 of the 12 480 Vac circuit breakers had current injection testing conducted without the full function test kit to verify the proper zone select interface jumper installation and proper breaker performance. Specifically, the 480 Vac load center main breaker 1B4A and the bus tie breaker BT-1B4A were not tested in accordance with corrective action item CR 2011-06621-32 prior to the action being closed. The team informed the licensee of this issue and the licensee initiated CR 2013-13262 to capture this in the station's corrective action program.

The licensee determined that Work Orders WO461130 and WO461131 were planned to conduct 480 Vac 1B4A load center breaker current injection testing without the full function test kit, but the work was not completed, and the corrective action item was incorrectly closed as completed. On July 7, 2013, the licensee performed current injection testing without the full functional test kit on main breaker 1B4A and the bus tie

breaker BT-1B4A to verify zone select interface jumpers were properly installed and proper breaker performance.

The team determined that the apparent cause of this finding was that the licensee failed to use conservative assumptions and conduct effectiveness reviews to validate injection testing without the full functional test kit was completed for all twelve 480 VAC circuit breakers prior to closing corrective action item CR 2011-06621-32.

Analysis. The licensee's failure to promptly identify and correct a condition adverse to quality is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone, and affected the associated objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 4, "PWR Refueling Operation: RCS level >23' or PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, and determined that the finding is of very low safety significance (Green) because the finding did not require a quantitative risk assessment because adequate mitigating equipment remained available. The finding had a cross-cutting aspect in the area of human performance associated with the decision-making component because the licensee did not ensure that the proposed action was safe in order to proceed, rather than unsafe in order to disapproved the action [H.1(b)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance's are promptly identified and corrected." Contrary to the above, an action to correct a condition adverse to quality was not completed when it was identified that injection testing with the full functional test kit would not verify proper zone select interface operation and proper breaker performance. Specifically, from January 15, 2013 to July 7, 2013, the licensee failed to conduct injection testing without the full functional test kit for the 480 Vac load center main breaker 1B4A and bus tie breaker BT-1B4A. On July 7, 2013, the licensee conducted injection testing without the full functional test kit for main breaker 1B4A and tie bus breaker BT-1B4A to verify proper zone select interface jumper installation and proper breaker performance. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CR 2013-13262, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-01, "Failure to Complete all Testing for a Condition Adverse to Quality."

(2) Introduction. The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records," associated with the failure to furnish evidence of an activity affecting quality associated with the 480 V breakers.

Description. On June 7, 2011, a fire occurred in the west switchgear room that caused extensive damage to 480 Vac switchgear 1B4A and associated equipment. The root cause of the fire was determined to be, "The design process failed to identify critical parameters and interfaces such as the silver plating contact area on the switchgear cubicle stabs," during a prior breaker replacement. One of the contributing causes to the fire was determined to be, "The design change specifications did not consider the partial plating of the switchgear stabs, resulting in the replacement breaker cradles engaging the bus stabs at the edge of and beyond the silver-plated contact area." Corrective Action 2 stated that the licensee would, "Re-align NLI breaker cradles so finger to bus stab engagement is in the silver plated contact surface, obtain acceptable as left digital low resistance ohmmeter (DLRO) readings under work orders…," and corrective Action 28 stated that the licensee would, "Develop a testing, inspection, and trending program to verify electrical connection adequacy. Use the resistance measurements obtained from the work order and trend the changes for appropriate adjustments to maintenance frequency and corrective actions."

During the team's review of the root cause analysis, they requested the basis for the licensee determining the DLRO values were acceptable. The licensee discovered that the engineering process for determining the acceptable DLRO values could not be found or identified because the individual who had provided the criteria had since retired. The licensee generated CR 2013-04032 to capture this concern in the station's corrective action program.

Corrective actions for CR 2013-04032 did not require the licensee to establish DLRO values for ensuring proper connections until the next refueling outage. The team questioned how the licensee was ensuring the DLRO measurements that were already taken were satisfactory and would ensure operability of the 480 Vac breakers. The licensee generated acceptance criteria to address this issue and reviewed the previously obtained DLRO values. Subsequently, during the review of previously obtained DLRO values the licensee found values outside the acceptance range. The licensee generated CRs 2013-14398 and 2013-14404 to capture this issue in the station's corrective action program.

Analysis. The licensee's failure to furnish evidence that showed the required DLRO values ensured proper connections between the Square D Masterpact breaker/cradle assemble to the GE AKD-5 480 V cubicle stabs was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it affected the design control attribute of the Mitigating Systems Cornerstone, and it directly affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate

safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding had a cross-cutting aspect in the area of human performance, associated with the resources component, because the licensee failed to maintain complete, accurate and up-to-date design documentation. Specifically, the licensee did not maintain the engineering process for determining acceptable DLRO values [H.2(c)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records," states, in part, that, "Sufficient records shall be maintained to furnish evidence of activities affecting quality…The records shall also include closely-related data such as qualifications of personnel, procedures and equipment…Records shall be identifiable and retrievable." Contrary to the above, from June 2011 through July 2013, the licensee did not maintain records related to the qualification of equipment in an identifiable and retrievable manner. Specifically, the licensee failed to maintain design documents that detailed the correct DLRO acceptance values required for ensuring proper connections between the Square D Masterpact NW breaker/cradle assemble to the GE AKD-5 480 Vac cubicle stabs. Because this finding is of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR 2013-04032, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000285/2013013-02, "Failure to Furnish Evidence of an Activity Affecting Quality."

(3) Introduction. The team identified a Severity Level IV violation of 10 CFR 50.59, "Changes, Tests, and Experiments," associated with the licensee's failure to adequately evaluate modification EC 33464, "Replace AK-50 480 V Main and Bus-Tie Breakers With Molded Case Type or Equivalent," to determine if it required prior NRC approval.

Description. In November 2009, the licensee implemented a modification to replace twelve General Electric AK-50 low voltage power circuit breakers with Nuclear Logistics Incorporated/Square-D Masterpact circuit breaker/cradle assemblies and digital trip devices. This modification was developed to address obsolescence issues and maintenance problems with the older AK-50 circuit breakers.

The licensee used General Electric AKD-5 Powermaster Low Voltage Drawout Switchgear, with a welded aluminum bus bar structure that transitioned to copper bus stabs in each breaker cell. The original AK-50 circuit breakers connected directly to the silver-plated areas on the line and load stabs. The new Nuclear Logistics Incorporated/Square-D circuit breaker design was an integrated unit consisting of a circuit breaker and cradle assembly. The cradle assembly converted the internal vertical breaker connectors to top and bottom spring-loaded horizontal finger assemblies which connected to the switchgear bus stabs.

Root Cause Analysis 2011-05414, which was performed to evaluate the June 7, 2011, fire in the 480 Vac Class 1E load center 1B4A, identified that the root cause of the fire was, "the design process failed to identify critical parameters and interfaces such as the

silver plating contact area on the switchgear cubicle stabs." It was determined that the finger assemblies extended beyond the silver-plated area on the switchgear bus stabs and interfaced directly with the copper portion of the stabs. The over extension of the finger assemblies, buildup of copper oxide, and residual hardened grease residue led to high resistance between the finger assemblies and stabs leading to the fire.

CR 2011-06319 was written after the fire for the discovery of the improper engagement of cradle fingers to silver plating on the stabs. The licensee re-analyzed the 50.59 that was completed as part of the initial breaker replacement modification (EC 33464). The team reviewed the licensee's implementation of the requirements in 10 CFR 50.59 for the modification. The team also reviewed the licensee's implementation of the requirements in Procedure FCSG-23, "10 CFR 50.59 Resource Manual," Revision 8, and Nuclear Energy Institute, "Guideline for 10 CFR 50.59 Implementation," (NEI-96-07), Revision 1. Procedure FCSG-23 is based on and incorporates the guidance in NEI 96-07.

The team noted that the screening process had determined that the finger assemblies' engagement with the stabs was not considered a credible failure mode, and that, it was stated that the Masterpact circuit breaker/cradle interface would not decrease the reliability of the equipment. The team recognized that this was in direct contradiction of the root cause documented in Root Cause Analysis 2011-05414, and that, if the licensee had properly implemented the requirements of 50.59 for the new credible failure mode associated with the finger assemblies engagement, the adverse impact would have required a 50.59 evaluation with the potential need for prior NRC review and approval.

In addition, the team identified that the new potential failure modes could have a significant impact regarding the reliability of the equipment. This is in contradiction with the NEI 96-07 screening criteria, which states that, "[t]he screening process is not concerned with the magnitude of adverse affects…." The qualifier which the licensee placed on the magnitude of the new potential failure modes may have resulted in the licensee missing other credible failure modes with adverse effects during the screening process.

The team informed the licensee of their concerns associated with the finger assemblies' engagement with the stabs not being considered a credible failure mode and the contradiction between the 50.59 screening and Root Cause Analysis 2011-05414. The teams also asked about the 50.59 screening using a significant decrease as the criteria for adverse effects instead of considering all/any adverse effects. The licensee entered this issue into their corrective action program as CRs 2013-04474 and 2013-16954. Based on the team's questions, the licensee has determined that a 50.59 evaluation was needed for modification EC 33464.

Analysis. The licensee's failure to implement the requirements of 10 CFR 50.59 and adequately evaluate changes associated with the electrical distribution system was a performance deficiency. Because this performance deficiency had the potential to impact the NRC's ability to perform its regulatory function, the team evaluated the performance deficiency using traditional enforcement. In accordance with

Section 2.1.3.E.6 of the NRC Enforcement Manual, the team evaluated this finding using the significance determination process to assess its significance. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design or qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its Technical Specification allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; (4) did not represent an actual loss of function of one or more nonTechnical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program; and (5) did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding, or severe weather event. Therefore, in accordance with Section 6.1.d.2 of the NRC Enforcement Policy, the team characterized this performance deficiency as a Severity Level IV violation. The team determined that a cross-cutting aspect was not applicable to this performance deficiency because the failure to adequately evaluate changes in accordance with 10 CFR 50.59 was strictly associated with a traditional enforcement violation.

Enforcement. Title 10 CFR 50.59, "Changes, Tests, and Experiments," Section (c)(1), states, in part, that a licensee may make changes in the facility as described in the Updated Safety Analysis Report without obtaining a license amendment pursuant to 10 CFR 50.90 only if: (1) a change to the Technical Specifications incorporated in the license is not required; and (2) the change, test, or experiment does not meet any of the criteria in Paragraph (c)(2). 10 CFR 50.59, Section (c)(2), states, in part, that a licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, if the change, would result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the Final Safety Analysis Report (as updated). Contrary to the above, the licensee failed to identify and evaluate new creditable failure modes to determine if they represented an adverse effect on the 480 Vac electrical distribution system, and therefore, did not perform the required 50.59 evaluation with the potential need for prior NRC review and approval. In addition, the licensee placed a qualifier on the magnitude of the adverse effects during the screening process, potentially missing other adverse effects introduced as part of modification EC 33464. The licensee's corrective action was to revise the evaluation. Because this violation was entered into the corrective action program as CRs 2013-04474, and 2013-16954, to ensure compliance was restored in a reasonable amount of time, and the violation was not repetitive or willful, this Severity Level IV violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2013013-03, "Failure to Evaluate Changes to Ensure They Did Not Require Prior Approval."

(4) Introduction. The team identified three examples of a Severity Level IV, non-cited violation of 10 CFR 50.73, "Immediate Notification Requirements for Operating Nuclear Power Reactors," associated with the licensee's failure to submit a licensee event report

within 60 days following a discovery of an event meeting the reportability criteria as specified.

Description. The team identified three examples of failure to make a required event notification within the 60 day time limit specified in 10 CFR 50.73.

Examples 1 and 2: The licensee failed to submit the required 60-day licensee event report for the 480 Vac 1B3A main breaker trip during the switchgear fault on 480 Vac 1B4A load center as required by: (1) Title 10 CFR 50.73(a)(2)(i)(B) for any operation or condition which was prohibited by the plant's Technical Specifications; and (2) 10 CFR 50.73(a)(2)(vii) for any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains. The licensee entered this issue into the corrective action program as CR 2013-12863.

Example 3: The licensee failed to submit the required 60-day licensee event report for a trip of the turbine-driven auxiliary feedwater pump following a start demand signal during a monthly operability surveillance test as required by 10 CFR 50.73(a)(2)(i)(B) for any operation or condition which was prohibited by the plant's Technical Specifications. The licensee entered this issue into the corrective action program as CR 2012-03796.

The team determined that, in both of these examples, the licensee had failed to thoroughly evaluate and identify all the associated reportability criteria for each issue.

Analysis. The team determined that the failure to make a required licensee event report was a violation of 10 CFR 50.73. The violation was evaluated using Section 2.2.4 of the NRC Enforcement Policy, because the failure to submit a required licensee event report may impact the ability of the NRC to perform its regulatory oversight function. As a result, this violation was evaluated using traditional enforcement. In accordance with Section 6.9 of the NRC Enforcement Policy, this violation was determined to be a Severity Level IV, non-cited violation. The team determined that a cross-cutting aspect was not applicable to this performance deficiency because the failure to make a required report was strictly associated with a traditional enforcement violation.

Enforcement. Title 10 CFR 50.73(a)(1) requires, in part, that licensees shall submit a licensee event report for any event of the type described in this paragraph within 60 days after the discovery of the event. Contrary to the above, between February 17, 2010 and June 20, 2013, the licensee failed to submit a licensee event report for three events meeting the requirements for reporting specified in 10 CFR 50.73. Because this violation has been entered into the corrective action program as CRs 2013-12863 and 2012-03796, compliance was restored in a reasonable amount of time, and the violation was not repetitive or willful, this Severity Level IV violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2013013-04, "Failure to Submit Licensee Event Report."

(5) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the licensee's approval of Root Cause

Analysis 2013-03424, Revision 0 and Revision 1, "MSPI Safety System Functional Failures Degrading Trend," which did not assure corrective actions to prevent repetition of a significant condition adverse to quality.

Description. The licensee approved Root Cause Analysis 2013-03424, Revision 0, "MSPI Safety System Functional Failures Degrading Trend," on July 8, 2013. This root cause analysis originally identified the root cause as, "Fort Calhoun Station's engineering management failed to maintain control over the design and configuration of Fort Calhoun Station." The corrective action to prevent recurrence in Root Cause Analysis 2013-03424, Revision 0, was documented as:

> "Identify and define the licensing bases and assure licensing bases documentation remains current, accurate, complete, and retrievable.
>
> - Identification includes determining the record types
> - Identify a consistent numbering system
> - Establish methodology (database) for ensuring current and historical licensing bases records are readily retrievable
> - Reconstitute (identify, locate, and store in a retrievable method) the licensing bases including historical records required to establish the current bases
> - If conflicts are identified during identification and location of licensing bases documentation, a condition report is initiated to document and track the resolution
> - Establish a process for assuring licensing bases documentation remains current, accurate, complete, and retrievable; current processes may be retained or revised to assure needed results
> - Closure determination: Conduct an outside independent assessment to validate the completion of identifying all license bases, documents are retrievable, and that the process for updates is implemented."

The team determined that the corrective action to prevent recurrence specified in Root Cause Analysis 2013-03424, Revision 0, was not appropriate and would not prevent recurrence of the root cause. The team determined that the root cause was narrowly focused on the management of the engineering division and failed to identify a culture in the engineering division, as a whole, that failed to maintain the design and configuration control. This licensee initiated CR 2013-12236 to place this issue in the station's corrective action program.

The licensee revised Root Cause Analysis 2013-03424 to include a new root cause and an additional corrective action. Root Cause Analysis 2013-03424, Revision 1 revised the root cause to, "Fort Calhoun Station failed to maintain an environment, in the Engineering Division, that valued maintaining the license and design basis of the station

over continued operation of the facility. This led to a loss of control over the design and configuration of Fort Calhoun Station." An additional corrective action to prevent recurrence was included to strengthen the function of the oversight group that performs reviews of engineering products.

During their review of Root Cause Analysis 2013-03424, Revision 1, "MSPI Safety System Functional Failures Degrading Trend," the team observed that Root Cause Analysis 2013-03424 extensively leveraged future actions associated with Root Cause Analysis 2013-05570, "Design and Licensing Bases Configuration Control," to (a) determine the extent of condition and extent of cause, (b) to effect corrective actions to preclude repetition, and (c) to complete the required effectiveness review. Consequently, the team examined the alignment between the two root cause analysis' and the licensee's quality-related corrective action program requirements to determine whether such cross-root cause analysis leveraging reasonably assured corrective actions to prevent repetition of significant conditions adverse to quality.

Although the closure review of Root Cause Analysis 2013-03424 would recognize its reliance on Root Cause Analysis 2013-05570, no requirement or process assured that the review would effectively evaluate changes to Root Cause Analysis 2013-05570 that could invalidate its tasked contribution to Root Cause Analysis 2013-03424. More importantly, although the corrective action program data system appeared capable of linking root cause analysis, no specific process was identified to ensure the assignments from Root Cause Analysis 2013-03424 would be recognized by the owner of Root Cause Analysis 2013-05570. In fact, the team confirmed there was no reference to Root Cause Analysis 2013-03424 in Root Cause Analysis 2013-05570 prior to the team's comments.

Further, the team determined that the use of future tasking to identify the extent of condition and extent of cause precluded the ability to assure that corrective actions approved to address the causes of the significant condition adverse to quality would be broad enough to prevent their repetition. In this specific instance, the significant condition adverse to quality or "Problem" was identified as the degradation of the Mitigating Systems Performance Indicator (MSPI) Safety System Functional Failure (SSFF) Performance Indicator (PI) to NRC White. The root cause was determined to be, the failure, "to maintain an environment, in the Engineering Division, that valued maintaining the license and design basis of the station over continued operation of the facility." The root cause analysis determined that, "other areas in the Engineering Division are susceptible to this cause," and they were not explicitly addressed in the root cause analysis. Likewise, the root cause analysis determined that, "loss of management oversight and control of programs has been shown to exist in the plant," and the degree of loss, and specific areas in which it has been identified, were not explicitly addressed in the root cause analysis. Rather these extent-of-cause determinations were largely future tasked to Root Cause Analysis 2013-05570.

Analysis. The licensee's failure to establish measures to assure that the cause of the degrading trend in MSPI safety system functional failures would be promptly identified and action taken to preclude repetition in accordance with 10 CFR Part 50, Appendix B,

Criterion XVI was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because the failure to correct the cause and preclude the repetition of the cause would have the potential to lead to a more significant safety concern. Specifically, failure to identify the correct cause and preclude repetition would lead to a high frequency of safety system functional failures. This finding was associated with the Mitigating Systems Cornerstone. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to be of very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the problem, and consequently, the resolution did not identify the extent of cause as necessary [P.1(c)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and 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, on July 8, 2013, measures established by the licensee failed to assure that the cause of an identified significant condition adverse to quality was corrected and corrective actions taken would preclude repetition. Specifically, measures established by the licensee failed to assure that the cause of an identified significant condition adverse to quality was corrected and corrective actions taken would preclude repetition involving a White mitigating system performance indicator associated with a degrading trend in safety system functional failures. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CRs 2013-584 and 2013-14614, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy:
NCV 05000285/2013013-05, "Inadequate Corrective Actions to Prevent Repetition of a Significant Condition Adverse to Quality, a White MSPI SSFF Degrading Trend."

(6) Introduction. The team identified multiple examples of a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to control deviations from design standards.

Description. In 2005, the licensee generated calculations and engineering documents needed to replace several reactor coolant system components, including the steam generators, pressurizer, reactor vessel head, and the associated structural supports. In

addition to upgrading the reactor coolant system components, the licensee also optimized the reactor coolant system support system with the removal of several structural supports and steel members. The team reviewed a small sample of the associated calculations and found several deficiencies where the station deviated from design basis requirements without a technical basis or justification.

In the first example, the team reviewed Design Calculation FC6945, "FCS RSG: RCS Structural Evaluation," and identified that the reactor coolant system piping stress levels exceeded the code allowable stress levels for accident loads. Specifically, the team found that reactor coolant system piping would exceed the allowable stress level for the faulted load combinations of an earthquake combined with a loss of coolant accident. In response to these concerns, the licensee performed an operability determination and generated CRs 2013-19878 and 2013-18361.

In the second example, the team reviewed Design Calculation FC7100, "Ft. Calhoun RCS Equipment Support Modifications due to NSSSRP," and Design Calculation FC7285, "Replacement Steam Generator (RSG) and Reactor Coolant Pump (RCP) Snubber Anchorage Upgrade Analysis," and identified that the reactor coolant system pipe supports credited concrete strength in excess of the design and licensing basis values. Specifically, the compressive strength of the concrete, per the design specifications and the Updated Safety Analysis Report, are 4000 psi or 5000 psi, depending on the location. However, the licensee used compressive strength values as high as 6000 psi in the calculations. The use of a higher compressive strength of concrete in the design calculations did not assure that appropriate quality standards are specified and included in design documents, and that, deviations from such standards are controlled. In response to this concern, the licensee generated CRs 2013-20281 and 2013-17885, and performed an operability determination. Using the design and licensing basis values, the anchor bolts were determined to be operable, but non-conforming.

In the third example, the team reviewed Design Calculations FC7100, FC7285, and FC6945, and identified that in several locations, the anchor bolts were designed to a lesser standard than required by the design and licensing basis. Specifically, the anchorage was designed to a safety factor of less than 4.0, as required by the licensing basis. The use of a lower safety factor for anchor bolts in the design calculations did not assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. In response to this concern, the licensee generated CRs 2013-14726 and 2013-20281. Using the design and licensing basis values, the anchor bolts were determined to be operable, but non-conforming.

Analysis. The failure to control deviations from quality standards as required by 10 CFR Part 50, Appendix B, Criterion III, was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the design control attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using

Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. There was no cross-cutting aspect assigned to this finding because this issue does not reflect present licensee performance.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design changes shall be subject to design control measures commensurate with those applied to the original design, which includes assuring that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, prior to December 5, 2013, the licensee failed to establish provisions to assure that deviations from specified quality standards were controlled. Specifically, the licensee failed to establish provisions to control the design of components within the reactor coolant system. The licensee took action to perform additional analysis to confirm the operability of the affected components and to determine the scope of the problem.

Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CRs 2013-19878, 2013-18361, 2013-20281, and 2013-14726, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-06, "Failure to Control Deviations From the Design Basis Requirements for Structural Calculations Related to the Reactor Coolant System."

(7) Introduction. The team identified multiple examples of a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee's failure to follow station procedures for corrective actions, operability, and calculation preparation for instances where the interim operability procedure was invoked for degraded conditions identified with piping and pipe supports. As a result, non-conservative design inputs were used without entering the non-conformances into the corrective action process or performing procedurally required operability evaluations.

Description. Station Procedure PED-QP-31, "Operability Determination Process," describes the licensee's operability determination process used by station personnel to assess the operability of structures, systems, and components (SSC) described in the licensee's Technical Specifications. The procedure defines degraded and nonconforming conditions as, "a condition of a SSC that involves a failure to meet the current licensing basis (CLB) or a situation in which quality has been reduced because of factors such as improper design….examples of nonconforming conditions include

when a SSC fails to conform to one or more applicable codes or standards (e.g., the CFR, operating license, Technical Specifications, Updated Safety Analysis Report, and/or license commitments)." Step 8.1 provides the licensee's requirement that operators immediately determine operability of degraded or nonconforming conditions:

> "Piping and pipe supports found to be degraded or nonconforming and that support SSC described in Technical Specifications should be subject to an operability determination."

Additionally, Station Procedure FCSG-24-1, "Condition Report Initiation," states, in part, that, engineering product errors that have been issued for implementation that would have had impact on the operation or qualification of a system or component, and errors in calculations, would require the initiation of a corrective action report.

The team reviewed Station Calculation FC07234, "Evaluation of Shutdown Cooling Mode Temperature and Pressure Increase on the Safety Injection System Piping and Pipe Supports," and found that the maximum deflection for certain elements of the shutdown cooling piping would exceed 1/8 inch. The Safety Evaluation Report for EA-FC-94-003 (dated April 16, 1993) requires an evaluation for deflection that exceeds 1/16 inch. However, the licensee accepted this condition as acceptable because it met PED-MEI-17, "Interim Operability Criteria," and engineering personnel considered the conditions acceptable without further review. The operations department was never informed of the degraded nonconforming condition.

The team reviewed Station Calculation FC02400, "Input Data Corresponding to Stress Summary RW-111A and Qualification Summary," Revision 5, and identified that the licensee used non-design criteria as acceptance criteria for multiple piping supports in the raw water system. Station Calculation FC02400, Revision 5, was not a restricted use analysis. The licensee explained that Revision 5 of Station Calculation FC02400 was a temporary analysis, not for full design use because it was marked as, "confirmation required," and such a marking restricted its use.

The team noted that Station Procedure PED-QP-3, "Calculation Preparation, Review and Approval," provides the requirement in Section 4.4.5 for restricting the use of a calculation:

"The use of unsubstantiated design inputs and assumptions in a calculation is permitted allowing the design process to proceed provided that they are identified as requiring confirmation (e.g., "Confirmation Required"). "Confirmation Required," is only used for inputs and assumptions which need to be substantiated at a later date, as determined by the calculation preparer. It shall not apply to the status of calculation methods (e.g., equations/computer codes). Confirmation shall be obtained before the modification has received a Multi-Discipline Independent Design Verification (IDV) or prior to the analysis becoming As-Built."

Analysis. The failure to provide adequate acceptance criteria for an activity affecting quality was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the human performance attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, and guidance from the Office of Nuclear Reactor Regulation, Division of Engineering technical staff for issues where the inputs to calculations deviated from approved standards, the finding was determined to have very low safety significance (Green) because: (1) the Office of Nuclear Reactor Regulation technical staff determined the non-conformances would not render the evaluated component as inoperable or unable to perform its safety function"; (2) it was not a deficiency affecting the design and qualification of a mitigating structure, system, or component; and (3) it did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of human performance associated with work practices component because the licensee failed to define and effectively communicate expectations regarding compliance with station procedures [H.4(b)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, prior to December 5, 2013, the licensee failed to complete activities affecting quality in accordance with prescribed procedures. Specifically, the licensee failed to recognize deviations from the design and licensing basis in engineering calculations were non-conforming conditions and follow the requirements of Station Procedure FCSG-24-1, "Condition Report Initiation," Station Procedure PED-QP-31, "Operability Determination Process," and Station Procedure PED-QP-3, "Calculation Preparation, Review and Approval," when invoking Station Procedure PED-MEI-17, "Interim Operability Criteria." The licensee's corrective action was to capture the identified instances in the corrective action program, and discontinue the use of the interim operability procedure. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CR 2013-03598, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-07, "Programmatic Failure to Evaluate Safety Impact of Degraded Conditions During Use of Interim Operability Criteria."

(8) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to correct conditions adverse to quality in safety-related equipment. The team identified multiple examples of this violation where an interim operability criteria procedure was applied instead of correcting the conditions adverse to quality.

Description. The team reviewed calculations FC06519 and FC06534 and found that the licensee identified that certain supports on seismic subsystem AC-215A stress levels exceeded design basis requirements, but failed to correct the condition adverse to quality.

Fort Calhoun Station, as part of a design basis reconstitution effort, reviewed several piping supports installed in the plant and performed analyses to confirm the as-installed configuration met the design basis requirements. In support of this effort, calculations FC06519 and FC06534 were originated on November 25, 1995 to analyze piping and piping supports that are a part of seismic subsystem AC-215A. Specifically, calculations FC06519 and FC06534 analyzed several supports for the raw water and component cooling water piping on the discharge lines of the containment air coolers. The calculations determined that the supports for seismic subsystem AC-215A would exceed the allowable stress specified by the design basis.

The team noted that the licensee had invoked Station Procedure PED-MEI-17, "Interim Operability Criteria," to determine that the supports were operable and were accepted as-is in the calculations. Corrective actions or configuration changes to restore the pipe supports in seismic subsystem AC-215A to acceptable stress levels specified by design basis requirements could not be found.

The team determined that the licensee had failed to promptly identify and correct conditions adverse to quality.

Analysis. The failure to correct conditions adverse to quality was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee had failed to implement a corrective action program with a low threshold for identifying issues to ensure that an issue potentially affecting nuclear safety are promptly identified and fully evaluated [P.1(a)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformance's are promptly identified and corrected." Contrary to the above, from November 25, 1995 to December 24, 2013, measures established by the licensee failed to assure that an identified condition adverse to quality was corrected. Specifically, the licensee failed to correct overstressed piping in the raw water system. The licensee's corrective actions included an extent of condition review to determine any other cases where "Interim Operability Criteria" was used but never addressed and developing a plan to correct the identified issues.

Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR 2013-22426, this violation is being treated as an non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-08, "Failure to Correct Overstressed Components."

(9) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to develop an adequate procedure for assessing operability.

Description. The team reviewed Station Procedure PED-MEI-17, "Interim Operability Criteria," which is a procedure the licensee used to evaluate critical quality equipment (CQE) and limited CQE piping and piping supports that are found to exceed design basis requirements. The procedure specifies specific criteria for evaluating the degraded piping and pipe supports to determine operability. The team identified a non-conservative equation used to calculate allowable bending stresses. The current equations listed in Station Procedure PED-MEI-17, Revision 2, do not comply with the requirements of ASME Section III, Subsection NF, for allowable bending stress criteria. Specifically, Station Procedure PED-MEI-17 only has one out of the two required criterion for bending stress. The procedure provides equations and criteria to increase allowable bending stress by a factor of two. However, an additional constraint is required by the ASME code. The second constraint is that the maximum allowable stress shall not exceed $0.7*S_u$ (70 percent of the ultimate strength of the material). Using the bending stress equations from Station Procedure PED-MEI-17 with common steel found in the plant would often make $0.7*S_u$ the limiting condition for allowable stress. Further, in certain cases the non-conservative stress criteria from Station Procedure PED-MEI-17 had the potential to allow structures to exceed their ultimate strength, but be within the allowable bending stress criteria found in the procedure.

The team determined that the licensee had invoked this procedure over 40 times since it was developed in 1990. Station Procedure PED-MEI-17 has been used to demonstrate operability on a large population of safety related structures, systems, and components, including the safety injection system, main steam system, feedwater system, steam generators, reactor coolant system, and raw water system.

The team informed the licensee of their concerns and the licensee initiated CR 2013-22342 to capture this concern in the station's corrective action program. Subsequently, the licensee determined that Station Procedure PED-MEI-17 was inadequate and suspended use of the procedure.

Analysis. The failure to use an adequate procedure for evaluating degraded or nonconforming pipe and pipe supports was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, and guidance from the Office of Nuclear Reactor Regulation, Division of Engineering technical staff for issues where the inputs to calculations deviated from approved standards, the finding was determined to have very low safety significance (Green) because: (1) the Office of Nuclear Reactor Regulation technical staff determined the non-conformances would not render the evaluated component as inoperable or unable to perform its safety function"; (2) it was not a deficiency affecting the design and qualification of a mitigating structure, system, or component; and (3) it did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. There was no cross-cutting aspect assigned to this finding because this issue does not reflect present licensee performance.

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and be accomplished, in accordance, with these instructions, procedures, or drawings. Contrary to the above, from May 3, 1990 to December 24, 2013, the licensee failed to provide a procedure appropriate for assessing operability for safety related piping and piping supports. The licensee's corrective action was to capture the identified instances in the corrective action program, and discontinue the use of the interim operability procedure. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CR 2013-22342, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-09, "Non-conservative Criteria in Operability Procedure."

(10) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to follow Station Procedure NOD-QP-31, "Operability Determination Process."

Description. CR 2012-09550 was written on August 17, 2012, to identify that components associated with Valve HCV-400F-O were beyond their currently documented service life. This represented a potential operability concern, and the

operability evaluation associated with this condition report determined that surveillance testing performed on May 29, 2011, provided a reasonable expectation that the valve was capable of performing its intended function.

In July 2013, during their review of the licensee's assessments of equipment service life issues, the team reviewed CR 2012-09550. The team determined that the documented operability evaluation did not provide a reasonable expectation of operability. Specifically, the surveillance testing the licensee had credited was a refueling surveillance and had an 18-month periodicity and was now outside of its specified periodicity and had not been performed since May 2011. Therefore, it no longer demonstrated operability for the degraded/nonconforming condition being evaluated.

The team informed the licensee of their concern with this valve, and asked if other components were crediting previously performed surveillance testing as a basis for operability. The licensee initiated CR 2013-12255 to capture this issue in the station's corrective action program.

The licensee subsequently determined that Valve HCV-400F-O had been repaired on April 30, 2013, and revised their operability evaluation to reflect this repair as the basis for operability of the component.

Analysis. The failure to properly assess and document the basis for operability, when a degraded or nonconforming condition was identified, was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Since the finding involved an inadequate operability determination while in a shutdown condition, the team used Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," and determined the finding to have very low safety significance (Green) because the finding did not increase the likelihood of a loss of reactor coolant system inventory, the finding did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory when needed, and the finding did not degrade the licensee's ability to recover decay heat removal once it was lost. This finding has a cross-cutting aspect in the area of human performance associated with the decision-making component because the licensee failed to use conservative assumptions in decision making when performing operability determinations [H.1(b)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Station Procedure NOD-QP-31, "Operability Determination Process," a procedure used to evaluate the operability of safety-related components, Step 4.3.15, required the licensee to properly assess and document the basis for operability when a degraded or nonconforming condition is identified. Contrary to the

above, on July 8, and July 15, 2013, the licensee failed to properly assess and document the basis for operability in accordance with prescribed procedures. The licensee addressed this issue by establishing an adequate basis for operability for the condition. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CRs 2013-15429 and 2013-14006, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-10, "Failure to Follow Operability Procedure."

(11) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to conduct an adequate evaluation of the impacts of modifying the turbine driven auxiliary feedwater pump (FW-10) during all modes of operation.

Description. The team noted during their review of NCV 05000285/2010006-01, "Failure to Correct Repeated Tripping of the Turbine-Driven Auxiliary Feedwater Pump FW-10," that the licensee had instituted an engineering change package to modify the turbine-driven Auxiliary Feedwater Pump FW-10, from a variable speed to a constant speed. The team reviewed the adequacy of this modification to ensure that the operation of this mitigating system component could still perform its intended function as required by the design and licensing basis.

The purpose of the auxiliary feedwater system is to provide an alternate source of feedwater to either or both steam generators in the event of a loss of main feedwater. The original design of the turbine-driven pump included a pneumatic loop controller which adjusted an actuator, determining the steam inlet throttle valve position (i.e. pump speed). There is also a mechanical speed-limiting governor which prevents the pump from damaging itself. Another protective feature of the pump is the backpressure trip device, which will close the throttle valve if sensed pressure in the steam outlet side is too high, again preventing pump damage.

The modification to change the pump from a variable speed to a constant speed setting was completed in 2009 as a corrective action for concerns regarding the reliability of the pneumatic speed control loop. It set the pump speed on the speed-limiting governor to approximately 7600 rpm (plus or minus 50 rpm); after the pneumatic loop control system was removed. The reasoning for this value was based on surveillance test data that indicated an average pump speed of 7550 rpm, which used a specific value for steam generator pressure. The pump would start and speed up until it reached the pre-set governor limit and then stay at the value until steam demand was decreased. This modification essentially resulted in the governor becoming the speed controlling device and the backpressure trip device acting as a protective measure if the governor were to fail. The engineering change package stated that, "it is not good practice to control a steam turbine's speed with a single device." However, the overpressure trip system is credited as backup.

While performing a review of Engineering Change Package 34435, "FW-10 Pneumatic Speed Control Removal," the team noted that the speed limiting governor for the pump

had been set to 7800 rpm previously. This value allowed FW-10 to provide a discharge pressure slightly higher than the anticipated peak steam generator pressures. It was also identified, through a review of design calculation models completed to look at a potential net positive suction head issue, that values of 7800 to 7900 rpm could be needed to deliver the required flow under certain scenarios and for specific steam generator pressures. Similar and higher pump speeds were identified as potentially being needed for specific scenarios analyzed while having only one steam generator available and for power uprate system upgrades.

New pump curves were not generated for FW-10 after the constant speed modification was made to analyze the wide variety of system pressures and flow requirements that could be needed and encountered during accident scenarios. The system changes could impact the safe operation of the governor or lead to a scenario where the pump would operate outside of the response of the governor when the pump was needed. Station Procedure PED-GEI-3, "Preparation of Modifications", Revision 91, Section 4.10.1, requires, in part, that system level functions shall be described in detail in the modification package, including modes of operation and methods of performing those functions, and all applicable performance and loading requirements shall be identified for each mode of operation. Also, Section 4.10.2, requires, in part, that all performance requirements, such as flow capacity, minimum temperature or pressure, and net positive suction head, shall be provided for each mode of operation.

<u>Analysis</u>. The failure to evaluate the effects of modifying the turbine driven auxiliary feedwater pump from a variable speed to a constant speed for all modes of operation was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the configuration control attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of human performance associated with the decision-making component because the licensee failed to use conservative assumptions in decision making. Specifically, the licensee did not reanalyze the pump performance parameters to identify any potentially adverse effects of changing the pump to a constant speed control [H.1(b)].

<u>Enforcement.</u> 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design changes shall be subject to design control measures commensurate

with those applied to the original design, which includes assuring that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, from 2009 through November 2013, the licensee failed to evaluate the effects of modifying the turbine driven auxiliary feedwater pump from a variable speed to a constant speed for all modes of operation. Specifically, the licensee did not reanalyze the pump performance parameters to determine whether any potentially adverse effects would occur from changing the pump to a constant speed when it is depended upon to mitigate accidents and respond appropriately to changes in operating conditions or design basis events. The licensee adequately addressed this issue by performing a detailed analysis which determined that the change did not adversely affect the function of the pump. Because the finding was of very low safety significance (Green) and has been entered into the station's corrective action program as CR 2013-10465, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-11, "Failure to Evaluate the Effects of Modifying the Turbine Driven Auxiliary Feedwater Pump."

(12) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's programmatic failure to conduct adequate operating experience reviews for root cause evaluations in accordance with Station Procedure FCSG-24-4, "Condition Report and Root Cause Evaluation," Revision 5.

Description. Station Procedure FCSG-24-4, "Condition Report and Root Cause Evaluation," states that the purpose of conducting an operating experience review is to determine whether the same or similar problems have occurred at the Fort Calhoun Station, and if, internal or industry operating experience was unsuccessful in preventing the problem. The procedure also states that an operating experience review shall be conducted in a systematic manner and both internal and external events from various sources shall be included.

A review of the problem statement to determine if the issue was a repeat event per the definition in the aforementioned procedure is also required. A repeat event is defined as a significance Level 'A' condition or event that shares the same or similar root causes as a previous event. Hence, there is a reasonable expectation that the event should not have occurred because a previous event's corrective actions to prevent recurrence should have prevented the event from occurring and, as such, it demonstrates that previous corrective actions to prevent recurrence were either ineffective or missing. If an issue is determined to be a repeat event then previous root cause corrective actions to prevent recurrence shall be reviewed to explain why they did not prevent the event, new corrective actions to prevent recurrence should consider why the previous corrective actions to prevent recurrence were not effective, and a condition report is generated describing the problem with the previous root cause(s).

The following were the specific examples associated with this performance deficiency:

1. During a review of the Equipment Service Life Root Cause Analysis (CR 2012-09491), it was noted by the team that zero failures were identified by the licensee from an Equipment and Information Exchange System (EPIX) search related to the failure of a 94/FSA relay, Model CR120B04022, which had failed on May 4, 1998. A review of industry operating experience by the team identified that there were failures related to this type of relay and that the average lifetime of this relay was 8 years. The team identified a discrepancy between this average lifespan and the site's assigned corrective actions to clean and inspect these relays every 10 years and to replace them every 20 years.

2. The team identified that the external operating experience search conducted for Root Cause Analysis 2012-08134, "Equipment Reliability," was limited to only Institute of Nuclear Power Operations (INPO) documents. FCSG-24-5 states that, "external operating experience includes, but is not limited to, EPIX, INPO website, vendor bulletins, 10 CFR Part 21 reports, NRC information notices, etc." The team noted that there were several NRC generic communications (e.g. Information Notices 2012-06 and 1993-64) related to equipment reliability that were missed in the operating experience review.

3. The team identified another example where the external operating experience search was incomplete. The Design and Licensing Bases Configuration Control Root Cause Analysis 2013-05570 external operating experience search omitted significant NRC operating experience (e.g. NUREG-1275, Volume 14, "Causes of Significance of Design-Basis Issues at U.S. Nuclear Power Plants," and NRC Information Notice 1998-40, "Design Deficiencies Can Lead to Reduced ECCS Pump Net Positive Suction Head During Design-Basis Accidents") as well as other external operating experience (e.g. licensee event reports from other plants related to several design issues including conducting a high energy line break analysis) that would aid the licensee in assigning corrective actions to prevent recurrence of the same problems.

4. When reviewing the operating experience section related to repeat events in CR-2013-5570 for the design and licensing basis root cause analysis the team identified that although the event was considered a repeat event it was not assessed in accordance with procedure requirements. Specifically, the questions posed in Station Procedure FCSG-24-4 that included why did previous corrective actions to prevent recurrence fail or previous root cause analyses not identify the issue, how will the new corrective actions to prevent recurrence fill in the gaps of the old ones, and issue a condition report to describe the missed opportunities with the previous corrective actions to prevent recurrence/root cause analyses, were not performed. The root cause analysis team stated that it was clear, by an operating experience review, that this issue was preventable but previous corrective actions to prevent recurrence were never written specific to this issue. The reasoning

for why previous corrective actions to prevent recurrence were never written or deficient was not evaluated nor were the operating experience opportunities that were missed and corrective actions/condition reports were not generated for these areas.

The team determined that this represented a programmatic failure by the licensee to conduct adequate operating experience reviews for root cause evaluations.

The team informed the licensee of their concerns and the licensee initiated CR 2013-14205 to capture this issue in the station's corrective action program for resolution.

Analysis. The licensee's programmatic failure to conduct adequate operating experience reviews for root cause evaluations was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because if left uncorrected it has the potential to lead to a more significant safety concern. Specifically, if the licensee does not thoroughly evaluate operating experience to determine whether the same or similar problems have occurred at the Fort Calhoun Station or within the industry, then effective corrective actions to prevent the issues from recurring may not be implemented and an adequate extent of condition and/or generic implications from the issue may not be identified. This finding was associated with the Mitigating Systems Cornerstone. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 4, "PWR Refueling Operation: RCS level >23' or PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, this finding was determined to be of very low safety significance (Green) because finding did not require a quantitative risk assessment because adequate mitigating equipment remained available. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the operating experience component because the licensee did not use operating experience information, including vendor recommendations and internally generated lessons learned, to support plant safety by implementing and institutionalizing operating experience through changes to station processes, procedures, equipment, and training programs [P.2(b)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, from December 2012 through August 2013, the licensee failed to complete activities affecting quality in accordance with prescribed procedures. Specifically, the licensee failed to follow the requirements of Station Procedure FCSG-24-4, and conduct adequate operating experience reviews during the performance of several root cause analyses, which could have prevented the identification and implementation of effective corrective actions to prevent recurrence. The programmatic aspect of this issue does not represent an immediate safety concern, and the licensee is developing corrective actions. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CR 2013-14205, this violation is being

treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-12, "Failure to Perform Adequate Operating Experience Reviews."

(13) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to fully incorporate applicable design requirements into the plant design.

Description. During reviews of the licensee's design documents, the team noted that the Fort Calhoun Final Safety Analysis Report and the Updated Safety Analysis Report both state that the vital switchgear rooms are cooled by a ventilation system that is capable of maintaining it below the operability requirements of the equipment under all conditions. However, the licensee had previously determined that the installed auxiliary building ventilation was not capable of maintaining the vital switchgear room's temperature under the design limits and had installed additional cooling units.

The team noted that the additional cooling units were not designated as safety-related components, and were not capable of functioning during all design events. Therefore, they were not capable of maintaining the room temperatures under all design requirements.

The team informed the licensee of their concerns and the licensee initiated CR 2013-09804 to capture this concern in the station's corrective action program.

The licensee determined that there was existing procedural guidance to open doors and provide temporary cooling to the vital switchgear rooms if the temperatures approached design limits or if ventilation was lost. Therefore, the licensee determined that a nonconforming condition existed, but that a reasonable expectation of operability existed based on the existing procedural guidance.

Analysis. The failure to fully incorporate applicable design requirements was a performance deficiency. The performance deficiency was determined to be more than minor, and therefore a finding, because it affected the design control attribute of the Mitigating Systems Cornerstone, and it directly affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of problem identification and resolution,

associated with the corrective action program component, because the licensee did not thoroughly evaluate the problem, and consequently, the resolution did not identify the extent of cause as necessary [P.1(c)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall be established to assure that applicable regulatory requirements and design bases, as defined in 10 CFR 50.2 and as specified in the license application, for those components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, from initial construction until present, measures established by the licensee did not assure that applicable regulatory requirements and design bases, as defined in 10 CFR 50.2 and as specified in the license application, for those components to which this appendix applies, were correctly translated into specifications, drawings, procedures, and instructions. Specifically, measures established by the licensee did not assure that the vital switchgear ventilation system was capable of maintaining the rooms' temperature below design requirements under all design requirements. This issue does not represent an immediate safety concern because the licensee has compensatory measures in place to maintain room temperatures, and the licensee is developing corrective actions to resolve this issue. Because this finding was of very low safety significance (Green) and has been entered into the corrective action program as CR 2013-9804, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-13, "Failure to Incorporate Design Requirements for Switchgear Room Cooling."

(14) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to take adequate corrective action regarding non-Category I (seismic) piping in the intake structure raw water vault.

Description. In a letter dated September 27, 1972, the Atomic Energy Commission (AEC) requested that the licensee determine whether the failure of any non-Category I equipment could result in flooding or release of chemicals that could jeopardize safe shutdown of the facility. The licensee was requested by letter, dated December 10, 1974, to determine whether the failure of any non-Category I equipment could result in a condition, such as flooding or the release of chemicals, that might affect the performance of safety related equipment required for safe shutdown of the facility or to limit the consequences of an accident. The circulating water (CW) and fire protection (FP) systems were required to be a part of this review. The licensee re-stated in a letter, dated February 14, 1975, that failure of the circulating water system does not affect safety related equipment. It did not appear to the team that the licensee evaluated piping or equipment in the intake structure. Based on the information provided by the utility, the NRC documented in an safety evaluation, dated February 18, 1978, that the existing plant design features provided sufficient protection from flooding which could result from the failure of non-Category I (seismic) system and are, therefore, acceptable.

NRC Inspection Report 05000285/89-50, dated February 20, 1990, documents multiple NRC concerns regarding loss of the raw water system. Specifically, due to the

configuration of the installation, the potential for common-mode failure of all four raw water pumps exists due to flooding vulnerabilities in the room. Leakage in the raw water header located inside the room or leakage from a system located above the room could cause the room to fill with water resulting in the loss of all four pumps. These design concerns were not previously reported to the NRC as discussed above.

A meeting was held as documented by NRC letter to OPPD dated April 9, 1990, to discuss these specific flooding concerns. OPPD indicated they would review these issues and identify appropriate corrective actions. Specifically, they would: (1) review internal flooding as an external event as part of the Individual Plant Examination / Probabilistic Risk Assessment analysis; (2) review occurrences outside the design basis and write a procedure to cover such an event; and (3) review the "critical crack" criterion and internal flood protection and address these items in the Updated Safety Analysis Report and/or design basis documentation.

EA90-084, "Raw Water Pump Room Internal Flooding," was developed using NRC Branch Technical Position MEB 3-1 criteria. Postulated failures of piping in the raw water pump rooms and of the Fire Protection piping above the pump rooms was evaluated to determine the potential for common-mode failure of all four raw water pumps. The analysis showed that for the worst-case credible water spray effects, fully applying branch technical position criteria results in possible scenarios for common-mode failure of all four raw water pumps from a single postulated pipe failure. The licensee states they are not committed to Branch Technical Position MEB 3-1.

The team raised a concern to the licensee regarding failure of non-Category I piping and potential effects on safety related equipment in the intake structure raw water vault. This concern was documented in CR 2013-05102. The team's specific concern regarding non-Category I circulating water piping running through the intake structure vault and the potential effects on the safety related raw water pumps was documented in CR 2013-10626.

The licensee contended that since EA90-084 analyzed the effects of ruptures from various sources the condition was acceptable. The team, however, noted that the station's current licensing basis did not allow for non-seismic interaction with safety related equipment, other than, as documented in the safety evaluation report issued by the NRC in 1978. Furthermore it did not appear that the licensee had reported these potential interaction concerns when originally requested by the Atomic Energy Commission.

Analysis. The failure to take adequate corrective action regarding non-Category I (seismic) piping in the intake structure raw water vault is a performance deficiency. The performance deficiency is more than minor, and therefore a finding, as it is associated with the design control attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated July 1, 2012, this finding was determined to have very low

safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. The finding has a cross-cutting aspect in the area of human performance associated with the decision-making component such that the licensee demonstrates that nuclear safety is an overriding priority. Specifically that the licensee uses conservative assumptions in decision making and adopts 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 Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Contrary to the above, from February 1975, through the present, the licensee failed to promptly identify and correct a condition adverse to quality associated with non-Category I (seismic) piping in the intake structure raw water vault. The licensee's corrective actions for this issue involved isolating and removing the piping. Because the finding was of very low safety significance (Green) and has been entered in the corrective action program as CRs 2013-04782, 2013-04956, 2013-09256, 2013-10626, and 2013-22090, this violation is being treated as an non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement policy: NCV 05000285/2013013-14, "Inadequate Corrective Action for Non-Seismic Category 1 Piping."

(15) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to follow Station Procedure NOD-QP-31, "Operability Determination Process," to adequately assess and document the basis for operability when a nonconforming condition was identified.

Description. CR 2013-13410 documents an NRC concern regarding seismic class I raw water piping in the non-seismic service building. The licensee's immediate operability determination concluded that the raw water system was operable but nonconforming due to being installed in a non-seismic building. The licensee determined that Abnormal Operating Procedure - 18, "Loss of Raw Water," provides guidance for the loss of raw water and would be used to mitigate the event. Therefore, it was an analyzed event.

The licensee failed to fully assess and document the basis for operability as required by Station Procedure NOD-QP-31. Specifically, the licensee did not determine the effect of a ruptured 6 inch stub in the raw water system with respect to the safety function provided by the raw water system during a design seismic event. The raw water system

function, during a seismic event, is provided in licensee analysis EA-93-085. This EA was not discussed in the immediate operability determination.

Additionally, the team determined that the licensee's position that having procedures that mitigate a loss of safety function implies that loosing that particular function has been analyzed was not correct. Specifically, while the loss of raw water procedure includes actions to implement in the event that all raw water is lost, this does not mean that the loss of raw water is within the current licensing basis.

<u>Analysis</u>. The failure to adequately assess and document the basis for operability regarding seismic raw water piping potentially interacting with the non-seismic service building is a performance deficiency. The performance deficiency is more than minor, and therefore a finding, as it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated July 1, 2012, this finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the problem such that the resolutions address causes and extent of conditions. This includes properly classifying, prioritizing, and evaluating for operability and report ability conditions adverse to quality [P.1(c)].

<u>Enforcement</u>. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Station Procedure NOD-QP-31, "Operability Determination Process," a procedure that is appropriate to the circumstances of evaluating the operability of safety-related components, Step 4.3.15, required the licensee to properly assess and document the basis for operability when a degraded or nonconforming condition is identified. Contrary to the above, on July 8, 2013, the licensee failed to complete activities affecting quality in accordance with prescribed procedures. The licensee revised the operability evaluation and established a reasonable basis for operability. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CRs 2013-13410 and 2013-13634, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of

the NRC Enforcement Policy: NCV 05000285/2013013-15, "Lack of an Adequate Operability Evaluation for Class 1 Raw Water Piping in Non-Class 1 Service Building."

(16) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," involving the licensee's failure to follow procedures when evaluating the impact of the removal of the motor for raw water Pump B on the intake cell level control during a potential site flood.

Description. The licensee performed an operability determination for Corrective Action 018 for CR 2011-10302. The operability determination was to evaluate the operability of plant equipment related to the classification of the intake structure river sluice gates as non-safety Class III components during the time it would take the NRC staff to review a license amendment request. This license amendment request would change the method of intake cell level control during a site flood from throttling the river sluice gates to use of the modified trash rack blowdown piping in the circulating water system.

The team reviewed the operability evaluation, and with consultation with the staff of the Office of Nuclear Reactor Regulation, and concluded the approach used by the licensee was acceptable until the license amendment was approved. The team further reviewed the operability determination for its consistency to actual plant configuration. In their review, the team identified under Section VII, "Justification of Decision," that the licensee noted that raw water Pump AC-10C, would not be available during a flood because it had a damaged cable jacket that would allow water intrusion into the cable. The team recalled from a recent plant walkdown that raw water Pump AC-10B was also unavailable at that time as it had its motor removed for refurbishment.

The team recalled from previous flooding inspections that the licensee's procedures could require two available raw water pumps for intake cell level control. With remaining raw water Pumps AC-10A and AC-10D available, the licensee met this procedural condition. The team noted that the procedure for flooding, Procedure AOP-01, "Acts of Nature," guided operators to run only one emergency diesel generator in an effort to meet a design requirement to maintain a 7-day fuel oil supply on site prior to a flooding event. Further inspection by the team revealed that raw water Pumps AC-10A and AC-10D could not be supplied by the same emergency diesel generator and hence to run raw water Pumps AC-10A and AC-10D, two emergency diesel generators would be required. Had a flooding event occurred at that time, the licensee could not have operated the plant within their design and procedures for raw water and diesel generator operations. Operators would have had to take on-the-spot actions outside their established procedures which would not have had the benefit of forethought to ensure other systems' design and qualifications were affected. The team did not find any discussion of this discrepancy in the operability determination.

The team determined that this was not in accordance with Procedure NOD-QP-31, "Operability Determination Process," Revision 44, which required that a positive determination of operability must be justified, including technical discussion of why the concern identified does not prevent the item from fulfilling its intended safety function.

The team considered the failure to address the design conflict in the operability determination to be a performance deficiency.

This issue did not represent an immediate safety concern and was entered into the licensee's corrective action program as CR 2013-15270.

The team noted that in September 2013, raw water Pump AC-10B was returned to service and the concern with the operability determination was no longer applicable.

Analysis. The failure to properly assess and document the basis for operability, when a degraded or nonconforming condition was identified, was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012, the finding was determined to have very low safety significance (Green) because it: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their Technical Specification allowed outage time; and (4) did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significance in accordance with the licensee's maintenance rule program. This finding has a cross-cutting aspect in the area of human performance associated with the work control component. Specifically, the team identified that the licensee failed to adequately plan and coordinate work activities in which interdepartmental coordination was necessary [H.3(b)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, on June 18, 2013, the licensee failed to complete activities affecting quality in accordance with prescribed procedures. Specifically, the operability determination for Corrective Action 018 for CR 2011-10302 was not performed in accordance with Procedure NOD-QP-31, "Operability Determination Process," Step 4.3.15, which required, in part, that, "A positive determination of operability must be justified, including…a technical discussion of why the concern identified does not prevent the item from fulfilling its intended safety function(s). This should demonstrate that the item is not exceeding its design basis specified in the reference documents." The licensee failed to evaluate the impact of having only two diversely powered available raw water pumps during a site flood on shutdown cooling system operability. The licensee addressed this issue by establishing an adequate basis for operability for the condition. Because the finding was of very low

safety significance (Green) and has been entered into the corrective action program as CR 2013-15270, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-16, "Inadequate Operability Determination Due to Failure to Consider an Unavailable Raw Water Pump."

(17) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to correctly translate the acceptance limit of intake sluice gate leakage values into procedures. Specifically, the acceptance limit from the licensee's testing was applied to 1000 feet of intake level and not to the 983 to 988 feet operating band prescribed in Section I – Flooding, of Procedure AOP-01, "Acts of Nature."

Description. The team reviewed Section I, "Flood," for Abnormal Operating Procedure AOP-01, "Acts of Nature," Revision 37, regarding the method and instructions for maintaining intake structure cell level. Abnormal Operating Procedure AOP-01 instructed operators in the "Instructions" or left-hand column of the procedure on how to accomplish the licensee's strategy. The team noted that, per Step 4.3.8G.2 of the Procedure FCSG-20, "Abnormal Operating Procedure and Emergency Operating Procedure Writer's Guide," Revision 10, that the expected or most likely conditions appear in the "Instructions" column. Procedure FCSG-20 further described that "Contingency Actions" in the right-hand column should contain guidance for exceptional circumstances, such as failing to meet an expected condition.

The team ascertained, from review of the right-hand or "Instructions" column, that the licensee's strategy to maintain intake cell level was to operate one raw water pump with all river sluice gates closed and throttle the four intake cell flood water inlet valves (CW-323, CW-324, CW-325, and CW-326), as necessary, to maintain cell level between 983 and 988 feet. Implicit in this strategy is that the leakage of the sluice gates would be within the capacity of the running raw water pump (or approximately 5325 gallons per minute) when cell level was in the 983 to 988 feet control band.

The licensee informed the team that sluice gate leakage had been monitored on May 11, 2013. The team reviewed the data from this leakage check which was Attachment 4 to the Operability Determination for Corrective Action 018 for CR 2011-10302. The team observed that, in this attachment, leakage had been measured and translated to a driving head (river level minus cell level) of 14 feet. The 14 feet value was noted on the attachment to provide additional margin in determining the acceptability of the in-leakage. On the attachment to the operability determination, which documented the testing, the sluice gate leakage was deemed acceptable if the leakage was within the capacity of one raw water pump with a 14 feet driving head for leakage. The team questioned the 14 feet value because a 14 feet driving head value would mean that the one running raw water pump could only keep up with the maximum acceptable sluice gate leakage at a cell level of 1000 feet during a design basis 1014 feet flood.

The team observed that the design calculation and testing for the station for intake cell level control was between 1000 and 1014 feet, yet the implementing procedure instructed operators to control between 983 and 988 feet. Additionally, based on the results of the May 11, 2013, testing which was within the 1000 feet leakage acceptance criterion, the licensee had set up a condition where implementation of their AOP-01 procedure for intake cell level control would make the Contingency Actions in the right-hand column part of the expected spectrum and not exceptional. This condition would be expected because the observed sluice gate leakage when translated to the 983-988 feet operating band would be in excess of the capacity of one raw water pump.

From this, the team concluded that the licensee had not properly translated the design of intake cell level control into the implementing procedures. The team determined that this failure was a performance deficiency.

Analysis. The failure to fully incorporate applicable design requirements was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the design control attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 4, "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, the team determined that because this finding did not increase the likelihood of a loss of reactor coolant system inventory; did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory, and did not degrade the licensee's ability to recover decay heat removal, this finding did not require a Phase 2 or 3 analysis as stated in Checklist 4. Therefore, the finding is determined to have very low safety significance (Green). This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate problems such that the resolutions address causes and extent of conditions [P.1(c)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall be established to assure that applicable regulatory requirements and the design bases, as defined in 10 CFR 50.2 and as specified in the license application, for those components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, from May 10, 2013, to the present, measures established by the licensee did not assure that applicable regulatory requirements and the design bases, as defined in 10 CFR 50.2 and as specified in the license application, for those components to which this appendix applies, were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the acceptance limit from the licensee's testing was applied to 1000 feet of intake level and not to the 983 to 988 feet design operating band prescribed in Section I – Flooding, of Procedure AOP-01, "Acts of Nature." This issue did not represent an immediate safety concern. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as

CR 2013-15287, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-17, "Failure to Translate Design Sluice Gate Leakage Into Operating Procedures."

(18) Introduction. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to maintain an adequate procedure for site flooding.

Description. The team reviewed Procedure AOP-01, "Acts of Nature," Revision 37, Section I, "Flood." Procedure AOP-01, Section I, Step 9.g, directed operators to maintain intake cell level between 983 and 988 feet by adjusting the four intake cell flood water inlet valves (CW-323, CW-324, CW-325, and/or CW-326) which were recently installed on the trash rack blowdown piping as part of a permanent modification.

Step 9.g had a contingency action in the right hand column which contained a typographical error that led to it being numbered as Contingency Action Step 9.h. The team pointed out the typographical discrepancy, which was not in accordance with the licensee's abnormal operating procedure writing guidance. Step 9.h.1 detailed the contingency action to be taken if operators were unable to maintain cell level less than 988 feet.

The team noted that the need for enacting the contingency action for being unable to maintain cell level less than 988 feet was a plausible condition based on the most recent measurement by the licensee of sluice gate leakage. The team noted that on May 11, 2013, the licensee measured the sluice gate leakage to be 2277 gallons per minute. This measurement was made with a driving head (the difference between river level and cell level) of 3.36 feet. The team translated this leakage to the driving head for what would be expected under design flood conditions (1014 feet river level and a 983-988 feet control band) and determined leakage would be greater than 6000 gallons per minute. This value was greater than the capacity of one raw water pump which was the operating configuration prescribed earlier in the flooding procedure.

Since level would be expected to exceed 988 feet due to sluice gate leakage, operators would then close the four intake cell flood water inlet valves (CW-323, CW-324, CW-325, and CW-326). Contingency Action 9.h.1 would then have the operators close the isolation valves for the four intake cell flood water inlet valves (CW-327, CW-328, CW-329, and CW-330). The team concluded that since these valves would only serve to stop any flow through the trash rack blowdown piping and not the sluice gate leakage, intake cell level would still not be able to be maintained less than 988 feet and Contingency Action 9.h.2 would need to be employed.

Contingency Action 9.h.2 instructs operators to start additional raw water pumps until the water level starts to fall if cell level is not able to be maintained less than 988 feet. The team concluded that this was a viable strategy to lower water level, but questioned the lack of specificity, particularly in not delineating qualitative and quantitative acceptance criteria, in the procedure from that point on to ensure intake cell level would be adequately maintained. The team noted that a specific level band was not called out

and direction on how to maintain that level band was not called out (whether by starting and securing raw water pumps or operating the intake cell flood water inlet valves).

The team, therefore, considered the procedure to be inadequate. Operators placed in those conditions would have to make an on-the-spot decision on how to proceed without the benefit of appropriate procedural guidance.

Analysis. The licensee's failure to maintain an adequate procedure for maintaining intake cell level during a flood was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affected the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 4, "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, the finding is determined to have very low safety significance (Green) because: (1) the finding did not increase the likelihood of a loss of reactor coolant system inventory; (2) did not degrade the licensee's ability to terminate a leak path or add reactor coolant system inventory; and (3) did not degrade the licensee's ability to recover decay heat removal. This finding did not require a Phase 2 or 3 analysis as stated in Checklist 4. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate problems such that the resolutions address causes and extent of conditions [P.1(c)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, prior to June 20, 2013, the licensee failed to provide instructions, procedures, or drawings which included appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Specifically, the licensee failed to include criteria for instructing operators on how to proceed if steps taken to maintain intake cell level less than 988 feet were unsuccessful. This issue did not represent an immediate safety concern. Because the finding was of very low safety significance (Green) and has been entered into the corrective action program as CR 2013-15289, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/201313-18, "Inadequate Procedure for Intake Cell Level Control During a Flooding Event."

(19) Introduction. The team identified a Green, non-cited violation of License Condition 3.D, "Fire Protection Program," for the failure to translate Appendix R license exemptions into the fire protection program design. Specifically, the licensee failed to translate the exemption from 10 CFR Part 50, Appendix R Section III.G that was granted July 3, 1985, for the intake structure Fire Area 31 into a design that met those exemptions.

Description. The licensee's fire protection program was defined in the Updated Safety Analysis Report and NRC safety evaluation reports. Section 9.11.1 of the Updated Safety Analysis Report describes the fire protection system design basis and states, in part, that the design basis of the fire protection system includes commitments to 10 CFR Part 50, Appendix R, Sections III.G, III.J, and III.O. Section 9.11.4.5 of the Updated Safety Analysis Report documented that descriptions of plant design and construction features for the fire protection program were contained in the Fort Calhoun Station Fire Hazards Analysis and Safe Shutdown Analysis. FHA-EA97-001, "Fire Hazards Analysis (FHA) Manual," Revision 16, Section 8.2.5 stated, in part, that a fire in Fire Area 31, "cable for raw water Pump AC-10B (EB-67, EB-7309, EA-7306, and EB-7307) have been encased in a 2-inch thick Pyrocrete enclosure located above the circulating water pumps." In a letter, "Request for Exemptions from Various Requirements of 10 CFR Part 50, Appendix R, Fire Protection Program for Nuclear Facilities," dated August 30, 1983, under Section III, Fire Area 31, Part B, in "Exemption Request," the licensee states, in part, that the District request an exemption from the requirements of Section III.G of Appendix R. Specifically, exemption is requested from the requirements that one pump and associated cables be completely enclosed in a 1 hour fire barrier enclosure and that complete, area-wide fire detection and suppression systems be provided for Fire Area 31. In Section (1), of this same section, it states in part, "The components necessary for cold shutdown in this fire area are the raw water Pumps AC-10A, B, C, and D. Power cables EA66, EB67, EC68, and ED69 for these pumps are located in this area. A Pyrocrete enclosure has been installed (details of which were transmitted to the Commission with our July 9, 1979 submittal) to protect the cables for Pumps AC-10A and AC-10B from any credible fire." The intake structure has fire detectors but does not have automatic fire suppression, and therefore, does not meet the requirements of having both fire detection and automatic fire suppression. Therefore, the licensee applied for an exemption with the above described enclosure providing protection for both raw water Pumps AC10-A and AC-10B cables.

The NRC in its July 3, 1985 letter to the licensee (NRC-85-200), which references the August 30, 1983 letter, responded to the license exemption request. In the evaluation under Intake Structure and Pull Boxes (Fire Area 31) it states in part, "In the Intake Structure, if a fire were to occur at the raw water pumps, it would be detected in its initial stages by the existing fire detectors. The fire brigade would then be summoned and would affect fire extinguishment using manual hose stations or portable fire extinguishers. During the time delay, associated with the arrival of the fire brigade, two of the pumps would be shielded from the effects of the fire by the concrete wall. In addition, smoke and heat from the fire would be vented upward and away from the pumps. Therefore, a complete 1 hour fire-rated barrier is not necessary to provide reasonable assurance that at least two pumps will remain free of fire damage." Based upon the above evaluation, the staff concludes that the existing fire protection provides an equivalent level of safety to that achieved by compliance with Section III.G. Therefore, the licensee's request for exemption for the intake structure and pull boxes is granted.

During walk down of the intake structure and review of the cable/conduit routing drawings associated with the intake structure, inspectors observed that there are two

pull boxes associated with raw water Pump AC-10B that are enclosed in a pyrocrete barrier. The pull boxes associated with AC-10B are PB-94T, which contains cable EB67, the 4.16 kV motor lead cabling for AC-10B and PB-93T, which contains cables EB7307, EB7309, and EB7314 low voltage control and power leads for discharge and isolation valves associated with AC-10B. The pull boxes associated with AC-10A are PB-91T, which contains cable EA66, the 4.16 kV motor lead cabling for AC-10A and PB-92T. Only one pull box associated with raw water Pump AC-10A is enclosed in a pyrocrete barrier and that is PB-92T, which contains cables EA7302, EA7306, and EA7313 that are low voltage control and power leads for discharge and isolation valves associated with AC-10A. Pull Box 91T, which contains cable EA66, the 4.16 kV motor lead cabling for AC-10A is not enclosed in the pyrocrete barrier and is also not shown to be enclosed in the fire barrier on the drawings. The drawings and the in situ equipment conditions match but neither conforms to the license exemption conditions since the motor lead cables associated with AC-10A are not enclosed in the pyrocrete barrier, and therefore, are not protected as stated by OPPD in the August 30, 1983, "Request for Exemption," letter to the Commission.

<u>Analysis.</u> The failure to translate Appendix R license exemptions into the fire protection program design is a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the protection against external factors attribute of the Mitigating Systems Cornerstone and affected the associated objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," dated September 20, 2013, Step 1.3, the team determined that the reactor would have been able to reach and maintain cold shutdown, therefore, this finding was determined to have very low safety significance (Green). There was no cross-cutting aspect assigned to this finding because the original license exemption request and grant was over 3 years ago and this issue does not reflect present licensee performance.

<u>Enforcement</u>. License Condition 3.D, "Fire Protection Program," requires, in part, that the licensee implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report and as approved in NRC safety evaluation reports. Section 9.11.1 of the Updated Safety Analysis Report describes the fire protection system design basis and states, in part, that the design basis of the fire protection systems includes commitments to 10 CFR Part 50, Appendix R, Section III.G. Contrary to the above requirement, from July 1983 until present, the licensee failed to implement and maintain in effect all provisions of the approved Fire Protection Program, which included the exemption that was granted in July 1983. Specifically, the licensee failed to translate Appendix R exemptions into a fire protection program design that met the requirements of the exemptions granted. This issue did not represent an immediate safety concern. Because this violation was of very low safety significance (Green) and has been entered into the corrective action program as CR 2013-15021, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000285/2013013-19, "Failure to Translate Appendix R License Exemptions into the Plant's Fire Protection Program Design."

(20) Introduction. The team identified a cited Severity Level IV violation of 10 CFR 50.9, "Complete and Accurate Information," and an associated reactor oversight process finding (NCV 05000285/2013013-19, "Failure to Translate Appendix R License Exemptions into the Plants Fire Protection Program Design"), for the licensees' failure to provide information to the Commission that was complete and accurate in all material respects.

Description. On February 4, 2008, the licensee submitted a letter, "Request for Exemption from Requirements of 10 CFR Part 50, Appendix R, Section III.G.1.b for Fire Area 31 at the Fort Calhoun Station," to the Commission requesting an exemption from the requirements of 10 CFR Part 50, Appendix R, Section III.G.1.b for the intake structure (Fire Area 31).

This exemption request was meant to address an issue with a previous Appendix R exemption, and two non-cited violations regarding not having the procedures and materials available in order to make repairs to cold shutdown equipment within 72 hours. Specifically, NCV 05000285/2004003-03, "Failure to Provide Fire Protection Features for Components Important to Achieve and Maintain Cold Shutdown," and Example 3 of NCV 05000285/2005008-06, "Failure to Take Prompt Corrective Action for Fire Protection Program Deficiencies," were issued to the station and while reviewing NCV 05000285/2005008-06, the licensee determined that the facilities Safety Evaluation Report, dated July 3, 1985, "Exemption Requests for the Fort Calhoun Station, Unit NO. 1 10 CFR PART 50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," incorrectly referenced Section III.G.2 and subsequently, provided exemption from 10 CFR Part 50, Appendix R, Section III.G.2, for the cables at the intake structure building and at the auxiliary building pull boxes. The licensee noted that the requirements of 10 CFR Part 50, Appendix R, Section III.G.2, are for equipment necessary for hot shutdown and the raw water system is credited to support cold shutdown functions for post-fire safe shutdown analysis. Therefore, Section III.G.2 was not applicable to Fire Area 31, and an exemption request needed to be submitted to request exemption from the requirements of 10 CFR Part 50, Appendix R, Section III.G.1.b. in lieu of Section III.G.2.

The licensee subsequently requested exemption from 10 CFR Part 50, Appendix R, Section III.G.1.b and the 72-hour requirement to provide repair procedures and materials for cold shutdown capability for redundant cold shutdown components, noting that, "OPPD currently has an approved exemption for the cable configuration at the auxiliary building pull boxes and at the intake structure building. However, the cables between these locations are not specifically discussed in that exemption. Therefore, this exemption request is to specifically address the cables in the duct bank and manhole vaults that are routed between the pull boxes and the intake structure building."

In a teleconference on September 25, 2008, the NRC provided additional clarification to information that was being sought in review of the request for exemption. The NRC requested the licensee to, "confirm that the pyrocrete enclosures were in place to protect the cables for raw water Pumps AC-10A and AC-10B from fire in the intake structure

building.” This request was based upon information that was provided by OPPD in the August 31, 1983, letter to the Commission in OPPD’s original request for exemption from Appendix R requirements which stated that, “a pyrocrete enclosure has been installed (details of which were transmitted to the Commission with our July 9, 1979 submittal) to protect the cables for Pumps AC-10A and AC-10B from any credible fire.”

The verbal request was subsequently communicated to the licensee by email (ML083360264) as a Request for Additional Information (RAI). Request for Additional Information 3 stated:

> Clarify and confirm that the types of combustibles have not changed and total combustible loading in the intake structure building has not increased, and that there is no change in active and passive fire protection features as last described in your letter dated August 30, 1983. If there is a change in the types of combustibles or there is an increase in combustible load or change in fire protection features in the intake structure building, the staff requests that the OPPD provide details and a basis for why the change remains acceptable. Also confirm that the pyrocrete enclosure is in place to protect the cables for raw water Pumps AC-10A and AC-10B from fire in the intake structure building.

On October 13, 2008, the licensee submitted a letter, “Response to Request for Additional Information Concerning Exemption from Requirements of 10 CFR Part 50, Appendix R, Section III.G.1.b. for Fire Area 31 at the Fort Calhoun Station,” to respond to the request for additional information documented in ML083360264. The licensee’s response to Request for Additional Information 3 stated, in part:

> The pyrocrete enclosure remains in place to protect cables associated with AC-10A and AC-10B from a fire in the intake structure. This enclosure is inspected by a fire barrier surveillance test on an 18-month interval.

In a letter dated February 6, 2009, “Fort Calhoun Station, Unit NO.1 - Exemption From the Requirements of 10 CFR Part 50, Appendix R, Section III.G.1.b,” the NRC granted an exemption from the specific requirements of Section III. G.1.b of 10 CFR Part 50, Appendix R, for the Fort Calhoun Station based upon its review and evaluation of the information provided in the licensee’s exemption request and response to NRC staff request for additional information questions.

While performing a walk down of the intake structure the team observed that Pull Box 91T, which contains the 4.16 kV motor leads for Pump AC-10A, was not protected by a pyrocrete enclosure like the 4.16 kV motor leads for Pump AC-10B. Therefore, only raw water Pump AC-10B is protected from a fire in the intake structure.

Analysis. The failure to provide the NRC with complete and accurate information when responding to a request for additional information was a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, “Issue Screening,” Figure 1, dated September 7, 2012, the team determined that the failure to provide complete and accurate information was a performance deficiency that required evaluation under both

traditional enforcement and the reactor oversight program. The performance deficiency was determined to be more than minor because: (1) the information was considered material to the NRC's decision making process; and (2) it affected the equipment performance attribute of the Mitigating Systems Cornerstone with regard to availability, reliability, and capability of the raw water pumps to perform their safety function during a fire in the intake structure. Using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the team determined the finding to have very low safety significance (Green) because it only affected the ability to reach and maintain cold shutdown conditions. Under the traditional enforcement review, the team determined that in accordance with Section 6.9.c.1 of the NRC Enforcement Policy, this finding represented a Severity Level III violation. Specifically, the team determined that if this information had been completely and accurately provided, it would likely have caused the NRC to undertake a substantial further inquiry. The NRC takes the issue of complete and accurate license submittals very seriously. For this reason, the NRC considered citing this as a Severity Level III violation, as discussed in the Enforcement Policy, since the NRC had approved a licensing action based on the incorrect information. However, after consideration by NRC management, and with the approval of the Director of the Office of Enforcement, it was determined that a Severity Level IV cited violation was appropriate. This decision was based on the very low safety significance (Green) of the associated reactor oversight process finding (05000285/2013013-19). There was no cross-cutting aspect assigned to this finding because the inaccurate information was provided over three years ago and this issue does not reflect present licensee performance.

<u>Enforcement</u>. 10 CFR Part 50.9, "Completeness and Accuracy of Information," requires, in part, that information provided to the NRC by a licensee shall be complete and accurate in all material aspects. Contrary to the above, the licensee responded to an NRC request for additional information in a letter dated October 13, 2008, with information that was not complete and accurate in all material respects. Specifically, the licensee stated that the pyrocrete enclosure remains in place to protect the cables associated with AC-10A and AC-10B from a fire in the intake structure when, in fact, the motor lead cables associated with raw water Pump AC-10A are not enclosed in the pyrocrete enclosure. This violation was entered into the corrective action program as CR 2013-15021. VIO 05000285/2013013-20, "Failure to Provide Complete and Accurate Information to the NRC."

(21) <u>Introduction</u>. The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to document the extent of condition review for a number of root cause analyses in accordance with corrective action program procedures. Specifically, during the course of the inspection, the team identified four examples where the licensee did not follow Station Procedure FCSG-24-4, "Condition Report and Cause Evaluation," and as a result did not evaluate the extent to which the actual conditions existed with other plant processes, systems, equipment, or human performance related activities.

<u>Description</u>. The team identified several instances where the licensee did not follow the corrective action program Station Procedure FCSG-24-4, "Condition Report and Cause

Evaluations." Specifically, the team identified four root cause analyses where the licensee did not identify other applicable plant processes, systems, equipment, or human performance related activities where the actual condition of the problem statement could exist. Since the extent of condition review is supposed to identify further deficiencies, and since corrective actions shall be planned to resolve those additional deficiencies (in accordance with Station Procedure FCSG-24-4), the licensee did not enter them into the corrective action program to ensure timely correction.

The following is a summary of the identified performance deficiencies with the references to the specific sections of the report where the issues are further described.

1. In RCA 2013-05570, "Design and Licensing Bases Configuration Control," the licensee's extent of condition review did not provide sufficient in-depth analysis and did not list the processes encompassed by the design and licensing bases. The team noted that since other processes are significantly impacted by this problem, including them as part of the review would have generated corrective actions associated with each specific process. For instance, processes such as operability determination, 50.59 Reviews, configuration control (tagging), design, vendor modifications, work control, Surveillance program, preventive maintenance process, and nondestructive examination would be impacted by the licensee's failure to maintain adequate configuration control of the structures, systems, components or activities, in accordance with, 10 CFR Part 50, Appendix B.

2. In RCA 2013-02857, "HELB/EEQ Not in Accordance with 10 CFR 50.59," the same-same review, which is part of the extent of condition review, consisted of other engineering programs at Fort Calhoun Station that are required by the Code of Federal Regulations (CFR) to be maintained current. The team noted that the RCA included some of the programs that were required to be maintained per the CFR but did not include 10 CFR Part 50, Appendix B, programs such as Nuclear Oversight, Quality Control, or commercial grade dedication programs. Since these programs require significant engineering and technical reviews, and are CFR required programs that needs to be maintained current, these were programs that should have been incorporated into the extent of condition review.

3. In RCA 2013-01796, "Unanalyzed Small Bore Piping Supports RCA," the similar-similar review, which is part of the extent of condition review, consisted of safety and non-safety related large bore piping. The licensee stated that the reason for concluding, that there is no extent of condition, was that large bore piping at Fort Calhoun Station was designed by computer analysis and not the generic nomograph and "eyeball" method. Additionally, the licensee stated that this piping was verified by inspection in response to IEB 79-14. The team noted that, based on the errors identified in previous engineering assumptions and calculations, the issues identified in the area of design and licensing basis maintenance and corrective action program root cause, as well as issues documented regarding thermal and cyclical fatigue analysis on Class I and II

piping, that large bore piping would have also been impacted in the extent of condition review.

4. In RCA 2012-01947, "Containment Integrity Issues with Electrical Penetration Assemblies Containing Teflon," the licensee did not perform a timely extent of condition review. Specifically, the extent of condition review associated with containment electrical penetrations with Teflon was performed, but was delayed due to core reload priorities.

Analysis. The failure to follow the requirements of Station Procedure FCSG-24-4, when documenting extent of condition reviews in multiple root cause analyses, was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because if left uncorrected the failure to perform extent of condition reviews could lead to a more significant safety concern. Specifically, the failure to identify and address additional conditions adverse to quality in the extent of condition review, has the potential to lead to a failure to recognize potentially degraded and non-conforming equipment in a timely manner. This finding was associated with the Mitigating Systems Cornerstone. Using Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 4, "PWR Refueling Operation: RCS level >23' or PWR Shutdown Operation with Time to Boil > 2 hours and Inventory in the Pressurizer," dated May 25, 2004, the team determined that the finding was of very low safety significance (Green) because the finding did not require a quantitative risk assessment because adequate mitigating equipment remained available. The team determined the Green finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee failed to thoroughly evaluate problems such that the resolutions address the causes [P.1(c)].

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, in three instances in 2013 and one instance in 2012, the licensee failed to follow the corrective action program Station Procedure FCSG-24-4, "Condition Report and Cause Evaluations." Specifically, the team identified four instances where the licensee, during the extent of condition review, did not identify other applicable plant processes, systems, equipment, or human performance-related activities where the actual condition of the problem statement in the root cause analysis could exist. The licensee has entered these issues into their corrective action program under several condition reports as described in this report. Because this finding was determined to be of very low safety significance and has been entered into the licensee's corrective action program, this performance deficiency is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 5000285/2013013-21, "Failure to Perform Adequate Extent of Condition Reviews."

(22) Introduction. The team identified a unresolved item associated with Calculation FC07234, "Evaluation of Shutdown Cooling Mode Temperature and

Pressure Increase on the SI System Piping and Pipe Supports." Specifically, the team is concerned with the methods used in the calculation which utilize ASME Section III requirements, but the plant is licensed to USAS B31.7 1968. In addition, the station is licensed to use, "Alternate Seismic Criteria Methodologies (ASCM)," but additional evaluation is required if a support or anchor is displaced more than 1/16 of an inch per the SER issued by the NRC and Calculation FC07234 only performed an evaluation when displacement exceeded 1/8 of an inch, per criteria established by a vendor memorandum.

Description. The Fort Calhoun Station's original code of record for safety-related piping is USAS B31.7, "Nuclear Power Piping," 1968 Draft Edition. The licensee reclassified a number of systems and piping in the early 1990's. In addition, because of the reclassification of some Class I piping to Class II piping, fatigue analysis was not performed on some safety related systems. The licensee reconciled the code of construction for some safety related systems to newer ASME Section III code, which requires a fatigue analysis for Class II piping, but because the plant is licensed to USAS B31.7, no analysis was completed. The NRC issued a safety evaluation allowing the Fort Calhoun Station to utilize alternate seismic monitoring criteria (ASCM) but stated additional evaluation was required if a support is displaced more than 1/16 of an inch. This safety evaluation was issued in April 1993.

The team reviewed Station Calculation FC07234, "Evaluation of Shutdown Cooling Mode Temperature and Pressure Increase on the SI System Piping and Pipe Supports," and noted that a vendor had performed this calculation utilizing criteria that deviated from the ASCM acceptance criteria. Specifically, the vendor used one of their internal memoranda, dated May 1979, to accept support displacement not exceeding 1/8 of an inch without evaluating the deviation as required by the ASCM safety evaluation. In addition, Station Calculation FC07234 identified some piping support stress allowables that were exceeded and needed additional vendor evaluation, but the only vendor evaluation noted was an email, with no justification or explanation why the loading on the SI-1A/B pumps and nozzles were acceptable. There were additional supports that exceeded their stress allowables but no additional evaluation is noted in the calculation.

Additional information is required to determine if Station Calculation FC07234 is adequate and fully supports operability evaluations for SI-1A/B pumps (High Head Safety Injection) and nozzles, AC-4A/B heat exchanger (Shutdown Cooling Heat exchangers) supply lines, high pressure safety injection and accumulator discharge piping, and the wall penetration bellows shown on Drawing IC-189. In addition, the open question regarding Class I and II reclassification that occurred in the 1990's needs to be reviewed to ensure that the right classification is applied to the Class I systems and that all of the thermal fatigue analysis, that is required, is completed.

Additional NRC inspection is necessary to determine if Station Calculation FC07234 is adequate. The team considered this to be an unresolved item, URI 05000285/2013013-22, "Shutdown Cooling Piping and Pipe supports Calculation Has Incorrect Acceptance Criteria for Anchor Displacement."

4OA6 Meetings, Including Exit

Exit Meeting Summary

On September 20, 2013, the team presented the inspection results in an on-site debrief to Mr. Louis P. Cortopassi, Vice President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented.

On February 18, 2014, the team presented the inspection results by conference call to Mr. Terrance Simpkin, Manager, Site Regulatory Assurance, and other members of the licensee staff. The licensee acknowledged the issues presented.

The team acknowledged that some of materials examined during the inspection were considered proprietary and controlled accordingly.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Adams, Principle Engineer Design Engineering (Retired Supplemental Worker)
D. Bakalar, Manager, Site Security
W. Beck, Exelon, Quad Cities RAM
J. Bonsum, EPM
B. Cable, Nuclear Safety Culture Coordinator
C. Cameron, Supervisor Regulatory Compliance
J. Cate, Supervisor, Nuclear Engineering
L. Cortopassi, Site Vice President
D. Digiacinto, Senior Nuclear Design Engineer Electrical/I&C
M. Doghman, VP Energy Delivery
K. Erdman, Supervisor, Engineering Programs
M. Ferm, Manager, Site Performance Improvement
M. Frans, Manager, Engineering Programs
R. Gaston, Licensing Manager
M. Greeno, NRC Inspection Readiness Team Contractor
R. Hall, GNJ Recovery Director
J. Hansen, VP OPPD
W. Hansher, Supervisor, Nuclear Licensing
R. Haug, Senior Consultant
M. Hirschfeld, Senior Organization Development Consultant
K. Ihnen, Manager, Manager, Site Nuclear Oversight
R. Hugenroth, Supervisor, Nuclear Assessments
J. James, Manager, Outage
R. King, Director, Site Maintenance
K. Kingston, Chemistry Manager/Nuclear Safety Culture Advocate
J. Kuzela, Control Room Supervisor
J. Lindsey, Training Director
T. Maine, Manager, Radiation Protection
T. Masne, RPM
E. Matzke, Senior Licensing Engineer
J. McManis, Manager, Projects
S. Miller, Manager, Design Engineering
V. Naschansy, Director, Site Engineering
B. Obermeyer, Manager, CAP
P. O'Neil, Senior Consultant, NWI Consulting, Inc.
T. Orth, Director, Site Work Management
A. Pallas, Manager, Shift Operations
M. Prospero, Division Manager, Plant Operations
J. Rainey, Human Resources Business Partner
B. Rash, Recovery Lead
K. Root, Regulatory

R. Short, Manager, Recovery
T. Simpkin, Manager, Site Regulatory Assurance
M. Smith, Manager, Operations
S. Swanson, Operations Director
K. Wells, Nuclear Design Engineer Design Electrical/I&C
J. Wiegand, Manager, Operations Support
G. Wilhelmsen, Exelon Nuclear Partners
J. Zagata, Reliability Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285/2013013-20	NOV	Failure to Provide Complete and Accurate Information to the NRC (Section 4OA4)
05000285/2013013-22	URI	Shutdown Cooling Piping and Pipe Supports Calculation Has Incorrect Acceptance Criteria for Anchor Displacement (Section 4OA4)

Opened and Closed

05000285/2013013-01	NCV	Failure to Complete all Testing for a Condition Adverse to Quality (Section 4OA4)
05000285/2013013-02	NCV	Failure to Furnish Evidence of an Activity Affecting Quality(Section 4OA4)
05000285/2013013-03	NCV	Failure to Evaluate Changes to Ensure They Did Not Require Prior Approval (Section 4OA4)
05000285/2013013-04	NCV	Failure to Submit Licensee Event Report (Section 4OA4)
05000285/2013013-05	NCV	Inadequate Corrective Actions to Prevent Repetition of a Significant Condition Adverse to Quality, a White MSPI SSFF Degrading Trend (Section 4OA4)
05000285/2013013-06	NCV	Failure to Control Deviations From the Design Basis Requirements for Structural Calculations Related to the Reactor Coolant System (Section 4OA4)
05000285/2013013-07	NCV	Programmatic Failure to Evaluate Safety Impact of Degraded Conditions During Use of Interim Operability Criteria (Section 4OA4)
05000285/2013013-08	NCV	Failure to Correct Overstressed Components (Section 4OA4)
05000285/2013013-09	NCV	Non-conservative Criteria in Operability Procedure (Section 4OA4)
05000285/2013013-10	NCV	Failure to Follow Operability Procedure (Section 4OA4)

Opened and Closed

05000285/2013013-11	NCV	Failure to Evaluate the Effects of Modifying the Turbine Driven Auxiliary Feedwater Pump (Section 4OA4)
05000285/2013013-12	NCV	Failure to Perform Adequate Operating Experience Reviews(Section 4OA4)
05000285/2013013-13	NCV	Failure to Incorporate Design Requirements for Switchgear Room Cooling (Section 4OA4)
05000285/2013013-14	NCV	Inadequate Corrective Action for Non-Seismic Category 1 Piping (Section 4OA4)
05000285/2013013-15	NCV	Lack of an Adequate Operability Evaluation for Class 1 Raw Water Piping in Non-Class 1 Service Building (Section 4OA4)
05000285/2013013-16	NCV	Inadequate Operability Determination Due to Failure to Consider an Unavailable Raw Water Pump (Section 4OA4)
05000285/2013013-17	NCV	Failure to Translate Design Sluice Gate Leakage Into Operating Procedure (Section 4OA4)
05000285/2013013-18	NCV	Inadequate Procedure for Intake Cell Level Control During a Flooding Event (Section 4OA4)
05000285/2013013-19	NCV	Failure to Translate Appendix R license Exemptions into the Plant's Fire Protection Program Design (Section 4OA4)
05000285/2013013-21	NCV	Failure to Perform Adequate Extent of Condition Reviews (Section 4OA4)

Closed

05000285-2012-002-00	LER	Inadequate Qualifications for Containment Penetrations Renders Containment Inoperable
05000285-2012-006-00	LER	Operation of Component Cooling Pumps Outside of the Manufacturers Recommendation
05000285-2012010-01	LER	Fire Causes a Circuit Breaker to Open Outside Design Assumptions
05000285-2012-016-00	LER	Unanalyzed Charging System Socket Welds to the Reactor Coolant System
05000285-2012-018-00	LER	Containment Air Cooling Units Operated Outside of Technical Specification during Cycle 26
05000285-2013-006-00	LER	Low Pressure Safety Injection and Containment Spray Pumps Mechanical Seals
05000285/2012010-01	VIO	Failure to Ensure that the 480 Vac Electrical Power Distribution System Design Requirements were Implemented and Maintained

Closed

05000285-2012-002-00	LER	Inadequate Qualifications for Containment Penetrations Renders Containment Inoperable
05000285-2012-006-00	LER	Operation of Component Cooling Pumps Outside of the Manufacturers Recommendation
05000285/2012007-02	VIO	Failure to Maintain Command and Control Function During Fire Fighting Activities in the Protected Area
05000285/2012004-04	VIO	Failure to Ensure Breaker Coordination of 480 VAC Electrical Power Distribution System Was Maintained

LIST OF DOCUMENTS REVIEWED

Section 4OA4: IMC 0350 Inspection Activities (92702)

PROCEDURES

NUMBER	TITLE	REVISION / DATE
NOD-QP-28	Safety Enhancement Program	
PED-QP-13	Design Basis Document Control	
PB-1	Writer's Guide for Plant Level Design Basis Documents	
SG-1	Writers Guide for System Design Basis Documents	
QAM-12	Quality Assurance Audit Scheduling	
SO-G-21	Modification Control	
PAP	Procedure Administration Program	
NPM-1.00	Nuclear Safety	5
NPM 2.04	Establishing and Maintaining a Safety Conscious Working Environment	4
NPM 2.04	Establishing and Maintaining a Safety Conscious Working Environment	5
FCSG-62	Site Nuclear Safety Culture Process	5
TBD-EPIP-OSC-1A	Recognition Category A, Abnormal Rad Levels/Radiological Effluent	2
EPIP-EOF-6	Dose Assessment	46
PBD-19	Electrical Equipment Qualification Program	4
PED-QP-15	Electrical Equipment Qualification Program	12

PROCEDURES

NUMBER	TITLE	REVISION / DATE
00314218-01	Flow Path Verification of Auxiliary Feedwater System	December 11, 2009
IC-CP-01-1368	Calibration of Auxiliary Feedwater Pump FW-6 Flow Loop F-1368	13
IC-CP-01-1369	Calibration of Auxiliary Feedwater Pump FW-10 Flow Loop F-1369	10
OP-ST-AFW-3009	Auxiliary Feedwater Pump FW-6 Steam Isolation Valve, and Check Valve Tests	21
OP-ST-AFW-3011	Auxiliary Feedwater Pump FW-10 Steam Isolation Valve, and Check Valve Tests	14
AOP-30	Emergency Fill of Emergency Feedwater Storage Tank	11
MGT-12-10	Safety Conscious Work Environment Training Slides	September 2012
MGT-12-12	Safety Conscious Work Environment Training Slides	Fall 2012
SE-ST-FW-3002	Feedwater Check Valves FW-161 and FW-162 Reverse Flow Test	12a
SO-M-101	Maintenance Work Control	96
SO-O-25	Temporary Modification Control	81
NOD-QP-19	Cause Analysis Program	43
EM-PM-EX-1200	Inspection and Maintenance of Model AKD-5 Low Voltage Switchgear	17
EM-PM-EX-1201	Inspection and Maintenance of Model AKD-5 Low Voltage Switchgear 1B4A	0
EM-PM-EX-0201	NLI Masterpact NW Circuit Breaker Inspection	20
EM-RR-EX-0203	Receipt Inspection of 480-Volt Square D/NLI Masterpact Type NW/NT Breakers/Cradles	0
EM-CP-05-1B4A-1	Calibration of Component Cooling Water Pump AC-3B Circuit Breaker	14
EM-PM-EX-0205	NLI Masterpact NT Circuit Inspection	1
EM-CP-05-1B4A-2	Calibration Procedure	R10

PROCEDURES

NUMBER	TITLE	REVISION / DATE
EM-CP-05-1B4A-3	Calibration Procedure Calibration of the Auxiliary Building MCC-4A2 Feeder Breaker	R10
EM-CP-05-1B4A-4	Calibration Procedure Calibration of Condenser Vacuum Pump FW-8B Circuit Breaker	R13
EM-CP-05-1B4A-5	Calibration Procedure Calibration of Screen Wash Pump CW-3B Circuit Breaker	R11
EM-CP-05-1B4A-6	Calibration Procedure Calibration of the Security Building Panel MS Feeder Breaker Located in Cubicle 1B4A-6	R9
EM-CP-05-1B4A	Calibration of the Main Circuit Breaker Located in Cubicle 1B4A	14
EM-CP-05-BT-1B4A	Calibration of 480 VAC Tie Breaker Located in Cubicle BT-1B4A	12
ERPG-EAG-01	Engineering Recovery Process Guide - Engineering Assurance Group	0
PED-GEI-2	Preparation of Procurement Specifications	16
PED-GEI-3	Preparation of Modifications	87
PED-GEI-7	Specification of Post Modification Test Criteria	15
PED-GEI-28	Preparation of Construction Work Orders	28
PED-GEI-29	Preparation of Facility Changes	55
PED-GEI-35	Preparation of Minor Configuration Changes	66
PED-GEI-52	Preparation of Field Design Change Requests	13
PED-GEI-60	Preparation of Substitute Replacement Items	45
PED-EWP-9	Testing of Control Circuits	0
FCSG-24-2	Evidence Quarantining	2
FCSG-24-5	Cause Evaluation Manual	5
FCSG-24-4	Condition Report and Cause Evaluation	3
FCSG-24-4	Condition Report and Cause Evaluation	5
NOD-QP-19	Cause Analysis Program	43
EM-ST-EE-0005	Capacity Discharge Test for Station Battery No. 1 (EE-8A)	23,25

PROCEDURES

NUMBER	TITLE	REVISION / DATE
FCSG-24-1	Condition Report Initiation	3
FCSG-24-3	Condition Report Screening	6a
FCSG-24-4	Condition Report and Cause Evaluation	6a
FCSG-24-5	Cause Evaluation Manual	5
SO-R-2	Condition Reporting and Corrective action	53b
FCSG-65-7	Program Restart Readiness	1
FCSG-65-8	Department Restart Readiness	2
NOD-QP-3	10 CFR 50.59 and 10 CFR 72.48 Reviews	35
NOD-QP-31.5	Degraded and Non-Conforming Evaluation	0
NOD-QP-38	Employee Concerns	9
NOD-QP-38	Employee Concerns	10
NOD-QP-X	Resolution of Differing Opinions	0
OI-AFW-4	Operating Instruction Auxiliary Feedwater Startup and System Operation	78
OP-ST-CCW-3002	AC-3A Component Cooling Water Pump Inservice Test	22
OP-ST-AFW-0004	Surveillance Test Auxiliary Feedwater Pump FW-10 Operability Test	31
PED-GEI-3	Preparation of Modifications	91
SE-ST-CCW-3002	CCW Pump Baseline Curve Procedure	10
SO-G-21	Modification Control	96
SO-R-1	Reportability Evaluation Checklist	20
SO-G-23	Surveillance Test Program	59

ENGINEERING ANALYSIS

NUMBER	TITLE	REVISION
EA-FC-06-032	Environmental Parameters for Electrical Equipment Qualification	0
EA-FC-10-020	Electrical Equipment Qualification Radiation Dose Reconstitution Analysis	0

ENGINEERING ANALYSIS

NUMBER	TITLE	REVISION
EA-11-037	Summary of Design Basis Reconstitution for High Energy Line Break (HELB) Outside of Containment in Response to CR 2007-3407	0
EA-FC-08-023	Vortexing in Safety-Related Tanks	14
EA-12-024	Determination of Design Temperature for Elastomers in Valves HCV-107A and HCV-1108A	
ACASR 2012-08621	Apparent Cause Evaluation-potential Elastomer Failure During a design Basis Accident for Valves HCV-238, HCV-239, and HCV-240	1
EA-FC-12-005	Harsh-mild Environment Threshold Criteria	0
EA-FC-12-0125	Electrical Penetration Feedthrough Classification and Qualification of Non-EEQ Penetration Feedthroughs	0

CONDITION REPORTS

NUMBER				
2005-04735	2005-04735-003	2005-04735-014	2006-06036	2007-02622
2007-03407	2007-02554	2008-04611	2009-02197	2009-04327
2009-05356	2009-06233	2009-00905	2009-05912	2009-04579
2009-05780	2009-02308	2009-04569	2009-01611	2009-12442
2009-05270	2009-05439	2009-05541	2009-05170	2009-04860
2009-06371	2009-06424	2009-05269	2009-04552	2009-06234
2010-04492	2010-03723	2010-00199	2010-01704	2010-01403
2010-04668	2010-00813	2011-08951	2011-00451	2011-08238
2011-05777	2011-07654	2011-00334	2011-06910	2011-07306
2011-01719	2011-02860	2011-06344	2011-07816	2011-09924
2011-02400	2011-08019	2011-09384	2011-09855	2011-01941
2011-06621	2011-05414	2011-02069	2012-08129	2012-08131
2012-04900	2012-03057	2012-03701	2012-04484	2012-04681
2012-10935	2012-05926	2012-06246	2012-06514	2012-10625
2012-13416	2012-10941	2012-10953	2012-12175	2012-14747
2012-13417	2012-02539	2012-13418	2012-13334	2012-13419
2012-08133	2012-11806	2012-13420	2012-13421	2012-13243
2012-03967	2012-11816	2012-12067	2012-02580	2012-11805
2012-11804	2012-11941	2012-11986	2012-04452	2012-07902

CONDITION REPORTS

NUMBER

2012-11982	2012-04169	2012-04280	2012-04444	2012-04467
2012-04490	2012-04536	2012-04602	2012-04903	2012-03986-019
2012-04262	2012-04262-021	2012-04662	2012-04262-022	2012-04262-023
2012-18336	2012-04262-055	2012-04262-058	2012-18336-001	2012-03986
2012-12443	2012-08123	2012-18338	2012-04899	2012-12378
2012-17353	2012-08129	2012-08124	2012-00451	2012-09494
2012-09112	2012-17354	2012-17355	2012-04594	2012-08137
2012-12044	2012-07112	2012-08642	2012-09111	2012-08123
2012-12430	2012-12305	2012-11986	2012-11987	2012-11994
2012-17352	2012-11982	2012-04662	2012-17362	2012-17353
2012-17572	2012-18336	2012-17361	2012-12460	2012-12547
2012-08142	2012-05580	2012-18338	2012-03254	2012-03974
2012-01541	2012-01910	2012-02723	2012-05134	2012-05509
2012-04132	2012-04516	2012-04850	2012-06452	2012-008621
2012-05569	2012-05846	2012-01640	2012-13620	2012-13694
2012-08684	2012-13299	2012-13306	2012-14517	2012-14736
2012-13919	2012-14045	2012-14464	2012-15218	2012-15440
2012-14800	2012-15116	2012-15215	2012-15690	2012-15696
2012-15441	2012-15666	2012-15687	2012-15747	2012-15750
2012-15697	2012-15703	2012-15721	2012-15805	2012-15844
2012-15755	2012-15758	2012-15770	2012-16038	2012-16145
2012-16023	2012-16025	2012-16030	2012-8851	2012-20806
2012-16171	2012-15399	2012-15750	2012-02534	2012-02881
2012-02026	2012-02115	2012-02498	2012-03805	2012-08521
2012-02947	2012-03397	2012-03796	2012-08737	2012-09179
2012-08522	2012-08526	2012-08528	2012-10477	2012-11874
2012-09196	2012-09494	2012-10206	2012-14958	2012-15721
2012-16900	2012-17447	2012-17717	2012-18345	2012-18347
2012-18675	2012-18793	2012-19477	2012-19769	2012-20128
2013-03056	2013-04037	2013-04034	2013-00730	2013-02202
2013-04167	2013-04286	2013-04223	2013-04032	2013-04033
2013-01396	2013-02278	2013-02557	2013-04504	2013-05026
2013-02710	2013-04141	2013-04442	2013-02611	2013-04680
2013-04806	2013-05018	2013-05026	2013-04547	2013-06267

CONDITION REPORTS

NUMBER

2013-05515	2013-05569	2013-05693	2013-05276	2013-05668*
2013-10507	2013-04937	2013-05663*	2013-05018	2013-05497*
2013-04934*	2013-04518*	2013-00907	2013-05674	2013-04377*
2013-01186	2013-00195	2013-03529	2013-01073	2013-01143
2013-03866	2013-01187	2013-03943	2013-03639	2013-03798
2013-04163	2013-03928	2013-04288	2013-04001	2013-04126
2013-04635	2013-04186	2013-05191	2013-04416	2013-04627
2013-05501	2013-04748	2013-05630	2013-05205	2013-05230
2013-00187	2013-03242	2012-08130	2013-05570	2013-05026
2013-12498	2012-08675	2013-12498	2013-14475	2010-1375
2010-0813	2012-08134	2013-14466	2009-2306	2013-14458
2009-3437	2010-5140	2013-02944	2013-02953	2013-14390
2013-02948	2013-02980	2013-03024	2013-11497	2012-01947
2013-14596	2013-04746	2012-08137	2013-15119	2012-08134
2013-02260	2011-9702	2013-14095	2013-13181	2013-14398
2013-14401	2013-04509	2011-10213	2012-01503	2012-00739
2012-05855	2013-04032	2012-01351	2012-00108	2012-01217
2013-16954	2013-05518	2011-10213	2011-9856	2012-01803
2013-14474	2013-04574	2011-9811	2012-00174	2012-01921
2011-9917	2011-5414	2011-10024	2011-9425	2011-8868
2011-10296	2011-10344	2012-00160	2011-8333	2012-10217
2012-10218	2011-9566	2012-01922	2013-14201	2011-8238
2012-01271	2012-01765	2012-01760	2012-00030	2011-6621
2012-01768	2013-00563	2011-10260	2012-01017	2011-5569
2012-18641				

WORK ORDERS

NUMBER

0056822-01	0097154-01	0097241-01	00125729-01	00335376-01
00314285-01	00338706-01	00314218-01	00357868-01	00370608
0370376-01	00437003-01	443770-01	450313-01	450346-01

WORK ORDERS

NUMBER

450348-01	450350-01	450351-01	450352-01	450353-01
450355-01	450357-01	472447-01	CWO 181503	CWO 329995-39
CWO 419854-01	CWO 421870-01	CWO 421871-01		

ACTION REQUESTS

NUMBER

2770	9290	9359	10237	13509
14047	14052	14053	14078	14097
14133	31024	36796	42918	51966
51959	53806			

MR-FC

NUMBER

97-007

EC

NUMBER

41455	53257	33464	34435	48714

FCSG

NUMBER

38	24	24-1	24-10	24-12
24-2	24-4	24-5	24-6	24-6.1
24-7	24-8	24-8.1	24-9	62

TREND CODES

NUMBER

ADE	ADI	ADP	OAI	OCR

CALCULATIONS

NUMBER

08081	07078	07076	06969	06148
06642	07536	05302	05374	06282
08179	08169			

DRAWINGS

NUMBER

11405-M-121	FO-4446	FO-1005	EM-1368/1369	00357868-01
80055	11405-M-253	11405-M-252	11405-M-253	EM-1039
11405-E-98	GHDR11405-S-2	A-748, Sheet 1		

LERS

NUMBER

2011-005	2011-007	2012-007	2012-008	2012-009
2012-010	2012-011	2012-012	2012-013	2012-014
2012-015	1988-019	2011-010-01	2011-010	2012-018
2012-002				

RCAS

NUMBER

2011-5414

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
	10 CFR 50.59 Evaluation of Manual Operator Action to open valve FW-1360	
SDBD-AC-CCW-100	CCW Design Basis Document	
TDB260.0020	Instruction Manual for Installation, Operation And Maintenance of MSB, MSC, MSD, MSE Horizontal, Multi-Stage Pumps	
NPM-100	Nuclear Safety	

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
MGT0302	Safety Culture	
MGT12-10	Safety Conscious Work Environment	
NPM-2.04		
	Final Closure Book for Resource Management	
FC06148	Auxiliary Feedwater Storage Tank Required Capacity	
FC05007	Usable Capacity of Emergency Feedwater Storage Tank FW-19	
FC06537		
TS-FC-87-231B	Memo	October 30, 1987
EM-PM-EX-1200		
PG-PDS-1		
AA/SA-PDS-3		
ECP-PDS-3		
SPD-PDS-7		
FPD	Safety Conscious Work Environment	
	Organizational Effectiveness Recovery Index	
RIS 2005-18	Effective Processes for Problem Identification and Resolution	
	Operations Memo 2007-01	
SEP-10	Safety Enhancement Program	
SEP-21	Safety Enhancement Program	
SEP-65	Safety Enhancement Program	
	FCS PI Report	
	FCS QA Audit	
	Final Closure Book for the FPD associated with Nuclear Safety Culture	
	Corporate Nuclear Oversight (GOSP) Committee Charter	September 18, 2012
ECP-03	IACDP Problem Development Sheet	

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NUMBER	TITLE	REVISION / DATE
FCS-95003-IACPD-03	IACPD – FCS Performance Goals Assessment Performance Area	
FCS-95003-IACPD-08	IACPD – FCS Audits and Assessments Assessment Performance Area	
FCS-95003-IACPD-02	IACPD – FCS Significant Performance Deficiencies Assessment Performance Area	
Policy 3.06	Corporate Governance, Oversight, Support, and Perform (GOSP) Model of Fort Calhoun Station”	July 27, 2012
RA 2013-0454	Governance & Oversight Self-Assessment	
	Mapping Leadership Skills/Attributes to Nuclear Safety Culture Results	February 2013
	95003 Collective Evaluation Final Report	
	FCS Nuclear Safety Culture Monitoring Panel First Quarter 2012 Report	
	FCS Nuclear Safety Culture Monitoring Panel Fourth Quarter 2012 Report	
	FCS Nuclear Safety Culture Senior Leadership Team Third Quarter 2012 Report	
MGT 12-10	Safety Conscious Work Environment	September 2012
USAR Appendix G	Responses to 70 Criteria	22
MR-FC-79-190C	Post-Accident Main Steam High Range Radiation Monitor RM-064, Final Design Package	0
Reg. Guide 1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	4
NRC Bulletin 88-04	Loop Accuracy for AFW Pump FW-6 Flow Channel Loop F-1368, Response to CAR 94-044	April 27, 1994
NUREG-1482	Guidelines for Testing at Nuclear Power Plants	1
PED-SYE-94-0297	Revised Accuracy for FM-1368-2 on IC-CP-01-1368, Reference Memo PED-SYE-94-0297	May 26, 1994
Nuenergy, Attachment 9, Final	Support of CDBI Self-Assessment Activities	0
LIC-80-0083	Response to Bulletin 80-10, Contamination of Nonradioactive Systems	July 3, 1980

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NUMBER	TITLE	REVISION / DATE
NRC-83-0015	NRC Resident Inspection	January 20, 1983
NRC-83-0092	NRC Resident Inspection	March 25, 1983
NRC-83-0185	NRC Resident Inspection	June 14, 1983
LIC-84-065	Application for Amendment of Operating License	March 7, 1984
LIC-84-209	Amendment 81 to Facility Operating License	July 12, 1984
LIC-85-009	Environmental Qualification of Safety-Related Electrical Equipment	January 10, 1985
LIC-88-929	Updated Response To Bulletin 88-04	November 4, 1988
LIC-12-0142	Licensee Event Report LER 2012-017	0
USAR-Appendix M	Postulated High Energy Line Repture Outside the Containment	10
USAR-9.4	Auxiliary Feedwater System	
USAR-Appendix M	Postulated High Energy Line Rupture Outside Containment	12
USAR-14.14	Steam Generator Tube Rupture Accident	15
NRC Bulletin 80-10	Contamination of Nonradioactive System and Resulting Potential Unmonitored, Uncontrolled Release of Radioactivity to Environment	May 6, 1980
NRC-04-024	Safety Evaluation for the Fourth 10-Year Interval Inservice Inspection Program Plan, Fort Calhoun	March 1, 2004
ASME OM Code 1988	Code For Operation And Maintenance Of Nuclear Power Plants	
NCV 05000285/2010006-01	Failure to Correct Repeated Tripping of the Turbine-Driven Auxiliary Feedwater Pump FW-10	August 12, 2010
NCV 05000285/2010006-02	Failure to Verify That the Turbine-Driven Auxiliary Feedwater Pump exhaust Backpressure Trip Lever was Fully Latched	August 12, 2010
NCV 05000285/2010006-03	Failure to Vent Control Oil Following Maintenance Results in Failure of the Turbine-Driven Auxiliary Feedwater Pump to Start	August 12, 2010
RCA 2013-0813	Root Cause Analysis Steam Driven Auxiliary Feedwater Pump (FW-10) Tripped Off	April 23, 2010

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NUMBER	TITLE	REVISION / DATE
PLDBD-ME-11	Internal Missiles and High Energy Line Break	15
EC48714	Installation of FW-10 Manual Trip Latch Clamp FW-64-C	0
NCR 449	Non Conformance Report	
NCR 410	Nonconformance Report Project # 093-15901	
	Recovery Issue Meeting Minutes for 1.c Closure Book	December 17, 2012 and February 8, 2013
FCS 95003	Project RSSPA Key Attribute Review Final Report for EDS & HPSI,	October 15, 2012
ERPG-DNC/OPEVAL-01	Engineering Recovery Process Guide – Degraded Nonconforming Conditions and Operability Evaluations	4
OPPD-E-12-002	Project Study Report – Study to Ensure Acceptable Diesel Generator Performance During Non-DBA Loss of Offsite Power Scenarios	0
SE-PM-EX-1600	Preventive Maintenance Infrared Thermographic Surveys	July 29, 2010
	Safety Conscious Work Environment at Fort Calhoun Station Rout Cause	1
	Fort Calhoun Station Nuclear Safety Culture Focus Groups, Summary of Findings	January 2013
	Fort Calhoun Station Nuclear “Two C’s” Meetings, Summary of Findings	January 2013
	Fort Calhoun Safety Culture Composite Index	December 2012
	Fort Calhoun Station Independent Safety Culture Assessment, Conger & Elsea, Inc.	May 2012
	Weekly Leadership Alignment Meeting Slides	February 4, 2013
	Weekly Leadership Alignment Meeting Slides	February 11, 2013
	Fort Calhoun Safety Culture Composite Index	January 2013
	Safety Conscious Work Environment Fundamental Performance Deficiency Analysis	July 2012

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NUMBER	TITLE	REVISION / DATE
	Corporate Governance, Oversight, Support and Perform Model of Fort Calhoun Station	
	Leadership/Organizational Effectiveness CR 2012-08130 and Nuclear Safety Culture CR 2012-08129 Fundamental Performance Deficiency Analysis	July 2012
	Corrective Action Program CR 2012-08124 Fundamental Performance Deficiency Analysis	July 2012
	Security Self Assessment Report	August 2012
SDBD-FW-AFW-117	System Design Bases Document Auxiliary Feedwater	44
STM	Auxiliary Feedwater System Training Manual	37